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ADVISORY MATERIAL FOR THE REGULATIONS FOR THE SAFE TRANSPORT OF RADIOACTIVE MATERIAL (1996 Edition)
IAEA Safety Standards Series No. ST-2

FOREWORD

Since the first edition in 1961, the Regulations for the Safe Transport of Radioactive Material of the International Atomic Energy Agency (IAEA Regulations) have served as the basis of safety for the transport of radioactive material world-wide. The provisions of the IAEA Regulations have been adopted in national regulations by most of the Member States of the Agency. The international regulatory bodies having responsibility for the various modes of transport have also implemented the IAEA Regulations. The safety record since the inception and throughout several comprehensive revisions of the Regulations has demonstrated the efficacy both of the regulatory provisions and of the arrangements for ensuring compliance with them.

During the discussions leading to the first edition of the IAEA Regulations, it was realized that there was a need for a document to supplement the Regulations which could give information on individual provisions as to their purpose, their scientific background and how to apply them in practice. The scientific basis of the classification of radioisotopes for transport purposes, then in use, and the factors that have to be taken into account by competent authorities in approving package designs, were examples adduced in support of this concept at the time. In response, the Agency published Safety Series No. 7, entitled, in its first edition in 1961, "Notes on Certain Aspects of the Regulations".

As experience in applying the Regulations grew, it became increasingly evident that, while the provisions of the Regulations might be essentially clear and unambiguous, nevertheless they would often also be highly technical in nature and unavoidably complex. Moreover they intentionally state no more than 'what' must be achieved in relation to package characteristics and operational conditions in order to assure safety. They do not seek to prescribe 'how ' the user should comply; indeed the freedom to innovate and to develop new ways to ensure compliance is recognized as intrinsically desirable in such a technically-advanced field. An additional source of information on the Regulations, providing advice on 'how' to comply with them which could be augmented from time to time in the light of latest experience, was therefore provided by the Agency, initially in relation to the 1973 Edition of the Regulations. This was entitled "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material". It was designated Safety Series No. 37.

Up to the time of publication of the previous edition of the IAEA Regulations, in 1985, Safety Series No.37 had reached its Third Edition. Meanwhile, Safety Series No. 7, which embodied information on the scientific basis and rationale of the Regulations, had been retitled "Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material" and, embodying mainly information on the scientific basis and rationale of the Regulations, was in its Second Edition.

During the current regulatory revision, which culminated in 1996, the Agency's senior advisory body for transport, the Transport Safety Standards Advisory Committee (TRANSSAC) agreed that it would be a useful simplification at this time to combine the two documents previously known as Safety Series No. 7 and Safety Series No. 37 in a single text, to be known as "Advisory Material for the Safe Transport of Radioactive Material". This would have the advantage of consolidating supporting information on the Regulations in one place, eliminating any possible duplication. The advisory function of the present document has been made paramount. The inclusion of some explanatory material supports this function since a proper understanding of the background to the regulatory provisions helps users to interpret them correctly and to comply with them fully.

Thus the primary purpose of this document (henceforth referred to as the Advisory Material) is to provide advice to users on proven and acceptable ways of achieving and demonstrating compliance with the Regulations. It must be emphasized that, in seeking to do this, the text is not to be construed as being in any way prescriptive. It offers guidance on ways of complying. It does not try to lay down 'the only way' to comply with any specific provision.

It must also be emphasized that the Advisory Material is not a stand-alone document. It only has significance when used as a companion to the IAEA Safety Standards Series No. ST-1, "Regulations for the Safe Transport of Radioactive Materials (1996 Edition)". To facilitate cross-referral between it and the Regulations, each paragraph of the Advisory Material is numbered correspondingly to the paragraph of the Regulations to which it most directly relates. To distinguish paragraphs of the Advisory Material from those of the Regulations for reference purposes, Advisory Material paragraphs always have a numeral after the decimal point, even when only one subparagraph of text exists. Thus, for example, advice relating to paragraph 401. of the Regulations should be initially sought under paragraph 401.1 of the Advisory Material. Integral paragraph numbers which are cited in the text, either alone or accompanied by lower case letters in brackets, should be taken as identifying paragraphs of the Regulations if this is not actually specified.

Member States and International Organizations are invited to take note of this publication and to bring it to the attention of persons and organizations who make use of, or are subject to, the IAEA Regulations. Moreover, readers are encouraged to send, through their competent authority, any comments they may wish to make, including proposals for modifications, additions or deletions, to the International Atomic Energy Agency.

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SECTION I

INTRODUCTION

BACKGROUND

103.1. When making national or international shipments it is necessary to consult the specific regulations for that mode of transport or the specific requirements for the countries where the shipment will be made. While most of the major requirements are in agreement with these regulations there can be differences with respect to the assignment of responsibilities for carrying out specific actions. For air shipments the International Civil Aviation Organization's (ICAO) Technical Instructions for the Safe Transport of Dangerous Goods by Air and the International Air transport Association's (IATA) "Dangerous Goods Regulations" should be consulted with particular regard to the state and operator variations. For sea shipments the International Maritime Organization's (IMO) International Maritime Dangerous Goods (IMDG) Code should be consulted. Some countries have adopted the IAEA Regulations by reference while others have incorporated them into their national regulations with possibly some minor variations.

OBJECTIVE

104.1. In general the Regulations aim to provide a uniform and adequate level of safety which is commensurate with the inherent hazard presented by the radioactive material being transported. To the extent feasible, safety features are required to be built into the design of the package. By placing primary reliance on the package design and preparation, the need for any special actions during carriage, i.e., by the carrier, is minimized. The overall level of safety which is achieved can be considered as the summation of these respective contributions, i.e., overall safety equals safety due to design plus safety due to operational controls.

SCOPE

106.1. Transport includes carriage by a common carrier or by the owner or employee where the carriage is incidental to the use of the radioactive materials such as vehicles carrying radiography devices being driven to and from the operations site by the radiographer, vehicles carrying density measuring gauges being driven to and from the construction site and oil well logging vehicles carrying measuring devices containing radioactive materials and radioactive materials used in oil well injections.

107.1. The Regulations are not intended to be applied to movements of radioactive material that forms an integral part of a means of transport, such as depleted uranium counterweights or tritium exit signs used in aircraft.

107.2. Cardiac pacemakers are an example of radioactive material implanted within persons. Another example is the case of people or animals treated with radioisotopes. The treating physician or veterinarian should give appropriate radiological safety advice.

107.3. Consumer products are those items which are available for sale and use to the general public as the end user without further control or restriction. They may be devices such as smoke detectors, luminous dials, or ion generating tubes that contain small amounts of radioactive substances. It is emphasized that consumer products are only outside the scope of the Regulations after sale to the end user. Any transport between manufacturers, distributors, and retailers is within the scope of the Regulations. Any conveyance used for the transport of consumer products to the point of retail sale to a member of the public as an end user, must meet the applicable requirements of the Regulations. This is to reduce the possibility that the transport of large quantities of individually exempted consumer products might fall outside the scope of the Regulations.

107.4. The principles of exemption and their application to the transport of radioactive material are dealt with in para. 401.

107.5. The scope of the Regulations includes those natural materials or ores which form part of the nuclear fuel cycle or which will be processed in order to use their radioactive properties. The Regulations do not apply to other ores which may contain naturally occurring radionuclides, but whose usefulness does not lie in the fissile, fertile or radioactive properties of those nuclides, provided that the activity concentration does not exceed 10 times the exempt activity concentration values. Natural material and ores containing natural occurring radionuclides which are processed are also exempt from the Regulations (up to 10 times the exempt activity concentration values) where the physical and/or chemical processing is not for the purpose of extracting radionuclides, e.g., washed sands, tailings from alumina refining etc,. Were this not the case, the Regulations would have to be applied to enormous quantities of material that present a very low hazard. However, there are ores in nature where the activity concentration is much higher than the exemption values. The regular transport of these ores may require a consideration of radiation protection measures. Hence, a factor of 10 times the exemption values for activity concentration was chosen as providing an appropriate balance between the radiological protection concerns and the practical inconvenience of regulating large quantities of material with naturally occurring low activity concentration.

108.1. Although these Regulations provide for the requisite safety in transport without the need for specified routeing, the regulatory authorities in some Member States have imposed routeing requirements. In prescribing routes, normal and accident risks, both radiological and non-radiological, as well as demographic considerations should be taken into account. Policies embodied in the routeing restrictions should be based upon all factors which contribute to the overall risk in transporting radioactive material and not only on concerns for `worst case' scenarios, i.e., `low probability/high consequence' accidents. The authorities at the state, provincial or even local levels may be involved in routeing decisions. In these cases it is often necessary either to provide them with evaluations to assess alternative routes or to provide them with very simple methodologies which they can use.

108.2. Through a co-ordinated research programme of the IAEA, a computer based environmental impact code INTERTRAN [1] has been developed and is available for use by Member States. In spite of many uncertainties, including the use of generalized models and the difficulty of selecting appropriate input values for accident conditions, this code is useful for calculating and understanding, at least on a qualitative basis, the factors which are significant in determining the radiological impact from the transport of radioactive materials. These factors are the important aspects which should be considered in any routeing decision. For routeing decisions involving a single mode of transport, many simplifying assumptions can be made and common factors can be assigned which result in easy-to-use relative risk evaluation techniques.

108.3. The consignor may also be required to provide evidence that measures to meet the requirements of safeguards and physical protection associated with radioactive nuclear material shipments are also complied with.

109.1. See paras 506 and 507.

SECTION II

DEFINITIONS

A_1 and A_2

201.1. See Appendix I.

Approval

204.1. The approval requirements in the Regulations have been graded according to the hazards posed by the radioactive material to be transported. Approval is intended to provide assurance that the design meets the relevant requirements and that the controls required for safety are adequate for the country and for the circumstances of the shipment. Since transport operations and conditions vary between countries, application of the `multilateral approval' approach provides the opportunity for each competent authority to satisfy itself that the shipment is to be properly performed, with due account taken of any peculiar national conditions.

204.2. The concept of multilateral approval applies to transport as it is intended to occur. This means that only those competent authorities through whose jurisdiction the shipment is scheduled to be transported are involved in its approval. Unplanned deviations which occur during transport and which result in the shipment entering a country where the transport had not previously been approved would need to be handled individually. For this reason the definition of multilateral approval is limited to countries "through or into which the consignment is transported" and specifically excludes countries over which the shipment may be transported by aircraft. The countries that will be flown over are often not known until the aircraft is in the air and receives an air traffic control clearance. If an aircraft is scheduled to stop in a country, however, multilateral approval includes approval by the competent authority of that country.

204.3. Users of the Regulations should be aware that a Member State may require in its national regulations that an additional approval be given by its competent authority for Special Form radioactive material, Type B(U) and Type C packages where the design has already been approved in another country and which is to be used for domestic transport on its territory.

205.1. For unilateral approval it is believed that the Regulations take into account the transport conditions which may be encountered in any country. Consequently, only approval by the competent authority of the country of origin of the design is required.

Carrier

206.1. The term "person" includes a body corporate as well as an individual (see also the Basic Safety Standards [2] paras 210-214).

Competent authority

207.1. The competent authority is the organization defined by legislative or executive authority to act on behalf of a nation, or an international authority, in matters involving the transport of radioactive material. The legal framework of a country determines how a national competent authority is designated and is given the responsibility to ensure application of the Regulations. In some instances, authority over different aspects of the Regulations is assigned to different agencies, depending on the transport mode (air, road, rail sea or inland waterway), or the package and radioactive material contents (Excepted, Industrial, Type A, Type B, Type C, special form, low dispersible material, fissile material, uranium hexafluoride) or on the hazard associated with the material (radioactive or other dangerous properties). A national competent authority may in some cases delegate the approval of package designs and certain types of shipments to another organization having the necessary technical competence. National competent authorities also constitute the competent authorities referred to in any conventions or agreements on the transport of radioactive material to which the country adheres.

207.2. The competent authority should make the consignors, carriers, consignees and public aware of its identity and how it may be contacted. It is helpful to publish the organizational identity (department, administration, office, etc.), with a description of the duties and activities of the organization in question as well as detailed mailing address, telex numbers, and telephone numbers, etc.

207.3. The primary source of competent authority identifications is the List of National Competent Authorities for transport, which is published annually by the IAEA and is available on request. Each country should ensure that the listed information is current and accurate. The IAEA requests verification of this information annually and prompt responses by Member States will ensure the continued value of this list.

Compliance assurance

208.1. See paras 311.1 to 311.9.

Confinement system

209.1. The confinement system should be that part of a package necessary to maintain the fissile material in the configuration that was assumed in the criticality safety assessment for an individual package (see para. 678). The confinement system could be (1) an inner receptacle with defined dimensions, (2) an inner structure maintaining the outer dimension of a fuel assembly and any interstitial fixed poisons, or (3) a complete package such as a irradiated nuclear fuel package with no inner container. Note that the containment system (see para. 213) consists of packaging components only, whereas the confinement system consists of packaging components and the package contents. Although the confinement system may have the same boundary as the containment system, this is not always the case since the confinement system maintains criticality control whereas the containment system prevents leakage of radioactive material. Each competent authority must concur that the confinement system defined in the criticality safety assessment is appropriate for the package design for both damaged and undamaged configurations (see paras 678).

Containment system

213.1. The containment system can be the entire packaging, but, more frequently, it makes up a portion of the packaging. For example, in a Type A package the containment system may be considered to be the vial containing the radioactive contents. The vial, its enclosing lead pot shielding and fibreboard box make up the packaging. The containment system does not necessarily include the shielding. In the case of special form radioactive material and low dispersible radioactive material, the radioactive material may be part of the containment system (see para.640)

213.2. The leaktightness requirement for a containment system depends on the radiotoxicity of the radioactive contents; for example a Type B(U) package under accident conditions must have the release limited to a value of A_2 in the period of a week. This connection to the A_2 value means that for highly radiotoxic isotopes such as plutonium the allowable leak rate will be much lower than for low-enriched uranium. However, if fissile material is able to escape from the containment system under accident conditions, it must be demonstrated that the quantity that escapes is less than that assumed in the criticality safety assessment in applying para. 682(c) or the criticality safety assessment may need to be modified.

Contamination

214.1. Contamination includes two types of radioactivity on surfaces or embedded in surfaces, namely fixed contamination and non-fixed contamination. There is no definitive distinction between fixed and non-fixed contamination and various terms have been used to describe the distinction. For practical purposes a distinction is made between contamination which, during routine conditions of transport, remains in situ and, therefore, cannot provide hazards from ingestion, inhalation or spreading, and non-fixed contamination which can contribute to these hazards. The only hazard from fixed contamination is that due to external radiation exposure, whereas the primary hazards from non-fixed contamination are the potential for internal exposure from inhalation and ingestion as well as external exposure due to contamination of the skin should it be released from the surface. Under accident conditions fixed contamination can become non-fixed contamination.

214.2. Contamination below levels of 0.4 Bq/cm^2 for beta and gamma emitters and for low toxicity alpha emitters, or 0.04 Bq/cm^2 for all other alpha emitters, (see also para. 508) can only give rise to insignificant exposure through any of these pathways.

214.3. Any surface with levels of contamination lower than 0.4 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters, or 0.04 Bq/cm^2 for all other alpha emitters is considered as a non-contaminated surface in applying the Regulations. For instance, a non-radioactive solid object with levels of surface contamination lower than the above limits is out of the scope of the Regulations and no requirement is applicable for its transport.

215.1. See para. 214.

216.1. See para. 214.

Criticality safety index

218.1. The criticality safety index (CSI) is a new term defined for the first time in the 1996 Edition of the Regulations. The 1973 Edition of the Regulations used a “transport index” for radiological control and control of criticality safety of packages with fissile material by establishing an allowable number of packages. The 1985 Edition of the Regulations defined a “transport index” which with one number accommodated both radiological safety and criticality safety considerations as the controlling quantity. As the operational controls needed for radiological protection and for criticality safety are essentially independent, this Edition of the Regulations has separated the CSI from the transport index (TI) which is now defined (see para. 243) only for radiological protection control. This separation into two indices enables a clear recognition of the basis for operational control of a fissile package and eliminates potential unnecessary restrictions caused by the use of one index. However, with this new control, care should be taken not to confuse the “new TI” and the “old TI” used in the previous Edition of the Regulations and to assure that labeling for criticality safety (see para. 544) and criticality control for packages, overpacks, and freight containers containing fissile material is based solely on the CSI value.

218.2. The criticality safety index (CSI) is a number used to control criticality safety for a shipment and is obtained by dividing the number 50 by the value of "N" (see para.528). The accumulation of packages containing fissile material should be controlled in individual consignments (see paras 529 and 530); in conveyances, freight containers, and overpacks (see paras 566(d) and 567) and in-transit storage. To facilitate such control, the CSI is required to be displayed on a label (see paras 544 and 545) which is specifically designed to indicate the presence of fissile material in the case of packages, overpacks or freight containers where contents consist of fissile material not excepted under the provisions of para. 672.

Exclusive use

221.1. The special features of an `exclusive use' shipment are, by definition, first that a single consignor must make the shipment and, through arrangements with the carrier, must have sole use of the conveyance or large freight container; and second, that all initial, intermediate and final loading and unloading of the consignment is carried out only in strict accordance with directions from the consignor or consignee.

221.2. Since ordinary in-transit handling of the consignment under exclusive use will not occur, some of the requirements which apply to normal shipments can be relaxed. In view of the additional control which is exercised over exclusive use consignments, specific provisions have been made for them which allow:

- Use of a lower integrity industrial package type for LSA materials;
- Shipment of packages with radiation levels exceeding 2 mSv/h, (but not more than 10 mSv/h), at the surface or a transport index exceeding 10;
- Increase by a factor of two in the total number of criticality safety indexes for fissile material packages in a number of cases.

Many consignors find that it is advantageous to make the necessary arrangements with the carrier to provide transport under exclusive use so that the consignor can utilize one or more of the above provisions.

221.3. In the case of packaged LSA materials the Regulations take into account the controlled loading and unloading conditions which result from transport under exclusive use. The additional controls imposed under exclusive use are to be in accordance with instructions prepared by the consignor or consignee (both of whom have full information on the load and its potential hazards), allowing some reduction in packaging strength. Because the uncontrolled handling of the packages is eliminated under exclusive use, the conservatism which is embodied in the normal LSA packaging requirements regarding handling has been relaxed, but equivalent levels of safety are to be maintained.

221.4. Packages which may be handled during transport must necessarily have their allowable radiation levels limited to protect the workers handling them. The imposition of exclusive use conditions and the control of handling during transport helps to ensure that proper radiation protection measures are taken. By imposing restrictions and placing a limit on the allowable radiation levels around the vehicle, the allowable radiation level of the package may be increased without increasing significantly the hazard.

221.5. Since exclusive use controls effectively prevent the unauthorized addition of radioactive materials to a consignment and provide a high level of control over the consignment by the consignor, allowances have been made in the Regulations to authorize more fissile material packages than for ordinary consignments.

221.6. For exclusive use of a conveyance or of a large freight container, the sole use requirement and the sole control requirement are the determining factors. Although a vehicle may be used to transport only radioactive material, this does not automatically qualify the consignment as exclusive use. In order to meet the definition of exclusive use, the entire consignment has to originate from or be controlled by a single consignor. This excludes the practice of a carrier collecting consignments from several consignors in a single vehicle. Even though the carrier is consolidating the multiple consignments onto one vehicle, it is not in exclusive use because more than one consignor is involved. However, this does not preclude a carrier or consignee who is consolidating shipments from more than one source from taking on the responsibilities of the consignor for these shipments and from being so designated.

Fissile material

222.1. A fission chain is propagated by neutrons. Because a chain reaction is dependent on the behaviour of neutrons, fissile material is packaged and shipped under requirements designed to control neutron behaviour in a manner to maintain subcriticality and, thus, provide criticality safety in transport. In the Regulations the term fissile material is occasionally used to refer both to fissile nuclides and to material containing those nuclides. Provided that one is alert to the context in which it is used, any ambiguities can be resolved.

222.2. Most nuclides can be made to fission, but many can only be made to fission with difficulty using special equipment and controlled conditions. The distinguishing characteristic of the fissile nuclides named in the definition is that they are capable of supporting a self-sustaining thermal neutron (neutron energies less than approximately 0.3 eV) chain reaction by only the accumulation of sufficient mass. No other action, mechanism, or special condition is required. For example, plutonium-238 is no longer listed in the definition because, although it can be made to support a fast neutron chain reaction under stringent laboratory conditions, in the form in which it is encountered in transport it does not have this property. Plutonium-238 cannot under any circumstances support a chain reaction carried by thermal neutrons. It is, therefore, 'fissionable' rather than 'fissile'.

222.3. As indicated in the above paragraph, the basis used to select the nuclides defined as fissile material for the purposes of these regulations relies on the ease of accumulating sufficient mass for a potential criticality. Other actinides that have the potential for criticality are discussed in ANSI/ANS-8.15 [3] and subcritical mass limits are provided for isolated units of ^{237}Np, ^{238}Pu, ^{240}Pu, ^{242}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{247}Cm, ^{249} Cf, and ^{251}Cf. The predicted subcritical mass limits for these materials range from a few grams (^{251}Cf) to tens of kilograms. However, the lack of critical experiment data, limited knowledge of the behavior of these nuclides under different moderator and reflection conditions, and the uncertainty in the cross-section data for many of these nuclides require that adequate attention (and associated subcritical margin) be provided to operations where sufficient quantities of these nuclides might be present (or produced by decay before or during transport). Advice of the competent authority should be sought on the need and means of performing a criticality safety assessment per the requirements of paras 671 - 682 whenever significant quantities of these materials may be transported.

Freight container

223.1. The methods and systems employed in the trans-shipment of goods have undergone a transformation since about 1965; the freight container has largely taken the place of parcelled freight or general cargo which was formerly loaded individually. Packaged as well as unpackaged goods are loaded by the consignor into freight containers and are transported to the consignee without intermediate handling. In this manner, the risk of damage to packages is reduced, unpackaged goods are consolidated into conveniently handled units and transport economies are realized. In the case of large articles such as contaminated structural parts from nuclear power stations, the container may perform the function of the packaging as allowed under para. 627.

223.2. Freight containers are typically designed and tested in accordance with the standards of the International Organization for Standardisation (ISO) [4]. They should be approved and maintained in accordance with the International Convention for Safe Containers (CSC) [5] in order to facilitate international transport operations. If other freight containers are used, the competent authority should be consulted. It should be noted however that the testing prescribed in CSC is not equivalent to that prescribed in ISO 1496/1. For this reason the Regulations require the design standard to be ISO.

223.3. In addition, special rules may be specified by modal transport organizations. As an example, the International Maritime Dangerous Goods (IMDG) Code [6] contains the provisions for the transport by sea of dangerous goods including radioactive material.

Low dispersible radioactive material

225.1. The concept of low dispersible radioactive material applies only to qualification for exemption from the requirements for Type C packages in the air transport mode.

225.2. Low dispersible radioactive material has properties such that it will not give rise to significant potential releases or exposures. Even when subjected to high velocity impact, and thermal environments, only a limited fraction of the material will become airborne. Potential radiation exposure from inhalation of airborne material by persons in the vicinity of an accident would be very limited.

225.3. The low dispersible radioactive material criteria are derived in consistency with other safety criteria in the Regulations, as well as on the basis of established methods to demonstrate acceptable radiological consequences, without taking any credit for the remaining damaged Type B packaging under severe aircraft accident conditions.

225.4. Low dispersible radioactive material may be the radioactive material itself, in the form of an indispersible solid, or a high-integrity sealed capsule containing the radioactive material, in which the encapsulated material acts essentially as an indispersible solid. Powders and powder-like materials cannot qualify as low dispersible material.

Low specific activity material

226.1. The basic reason for the introduction of a category of low specific activity materials (LSA) into the IAEA Regulations was the existence of certain solid materials, the specific activities of which are so low that it is inconceivable that, under any circumstances arising in transport, a sufficient mass of such materials could be taken into the body to give rise to a significant radiation hazard. Uranium and thorium ores and their physical or chemical concentrates are materials falling into this category. This concept was extended to include other solid materials on the basis of a model which assumes that it is most unlikely that a person would remain in a dusty atmosphere long enough to inhale more than 10 mg of material. If the specific activity of the material is such that the mass intake is equivalent to the activity intake assumed to occur for a person involved in a median accident with a Type A package, namely 10^{-6} A_2, then this material should not present a greater hazard during transport than the quantities of radioactivity transported in Type A packages. This leads to a low specific activity limit of 10^{-4} A_2/g.

226.2. Consideration was given to the possibility of shipping solid objects without any packaging. The question arose for concrete blocks (containing radioactivity throughout the mass), for irradiated objects and for objects with fixed contamination. Under the condition that the specific activity is relatively low and remains in the object either by activation or on the object as fixed contamination on its surface, the object can be dealt with as a package. For the sake of consistency and safety, the radiation limits at the surface of the unpackaged object should not exceed the limits for packaged material. Therefore, it was considered that above the limits of surface radiation levels for packages (2 mSv/h for non-exclusive use and 10 mSv/h for exclusive use), the object must be packaged in a industrial package which assures shielding retention in routine transport. Similar arguments were made for establishing surface contamination levels for unpackaged surface contaminated objects (SCO).

226.3. The preamble to the LSA definition does not include the unshielded radiation level limit of 10 mSv/h at 3 m (para. 521) because it is a property of the quantity of material placed in a single package rather than a property of the material itself (although in the case of solid objects which cannot be divided, it is a property of the solid object).

226.4. The preamble also does not include wording relative to the essentially uniform distribution of the radionuclides throughout the LSA material. However it states clearly, that the material should be in such a form that an average specific activity can be meaningfully assigned to it. In considering actual materials shipped as LSA, it was decided that the degree of uniformity of the distribution should vary depending upon the LSA category. The degree of uniformity is thus specified, as necessary, for each LSA category (see, for example, para. 226(c)(i)).

226.5. LSA-I was introduced in the 1985 Edition of the Regulations to describe very low specific activity materials. These materials may be shipped unpackaged or they may be shipped in Industrial Packages Type 1 (Type IP-1) which are designed to minimal requirements (para. 621). According to para. 226(a) (i) LSA-I materials cannot consist of: concentrates of ores other than uranium or thorium concentrates (for example, radium ore concentrate cannot be LSA-I material), except if they meet para. 226 (a) (iv). In the 1996 Edition of the Regulations the LSA-I category was revised to take into account

- the clarification of the scope of the Regulations concerning ores other than uranium and thorium ores according to para. 107 (e),
- fissile materials in quantities excepted from the package requirements for fissile material according to para. 672, and
- the introduction of new exemption levels according to para. 236.

The definition of LSA-I was consequently modified

- to include only those ores containing naturally occurring radionuclides which are intended to be processed for the use of these radionuclides (para. 226 (a) (i)),
- to exclude fissile material in quantities not excepted under para. 672 (para. 226 (a) (iii)), and
- to add radioactive material in which the activity is distributed throughout in concentrations up to 30 times the exemption level (para. 226 (a) (iv)).

Materials containing radionuclides in concentrations above the exemption levels have to be regulated. It is reasonable that materials containing radionuclides up to 30 times the exemption level may be exempted from parts of the transport regulations and may be associated to the category of LSA-I materials. The factor of 30 has been selected to take account of the rounding procedure used in the derivation of the Basic Safety Standards [2] exemption levels and to give a reasonable assurance that the transport of such materials does not give rise to unacceptable doses.

226.6. Uranium enriched to 20% or less may be shipped as LSA-I material either in Type IP-I packages or unpackaged in fissile excepted quantities. However, amounts exceeding fissile-excepted quantities (see para. 672) will be subject to the requirements for packages containing fissile material, thus precluding transport of the material unpackaged, or in unapproved packages.

226.7. The materials expected to be transported as LSA-II could include nuclear reactor process wastes which are not solidified, such as lower activity resins and filter sludges, absorbed liquids and other similar materials from reactor operations, and similar materials from other fuel cycle operations. In addition, LSA-II could include many items of activated equipment from the decommissioning of nuclear plants. Since LSA-II materials could be available for human intake after an accident, the specific activity limit is based upon an assumed uptake by an individual of 10 mg. Since the LSA-II materials are recognized as being clearly not uniformly distributed (e.g., scintillation vials, hospital and biological wastes, and decommissioning wastes), the allowed specific activity is significantly lower than that of LSA-III. The factor of 20 lower allowed specific activity as compared to the limit for LSA-III compensates for localized concentration effects of the non-uniformly distributed material.

226.8. While some of the materials considered to be appropriate for inclusion in the LSA-III category would be regarded as essentially uniformly distributed (such as concentrated liquids in a concrete matrix), other materials such as solidified resins and cartridge filters are distributed throughout the matrix but are uniformly distributed to a lesser degree. The solidification of these materials as a monolithic solid which is insoluble in water and non-flammable makes it highly unlikely that any significant portion of it will become available for intake into a human body. The recommended standard is intended to specify the lesser degree of activity distribution.

226.9. The provisions for LSA-III are intended principally to accommodate certain types of radioactive waste consignments with an average estimated specific activity exceeding the 10^{-4} A_2/g limit for LSA-II materials. The higher specific activity limit of 2×10^{-3} A_2/g for LSA-III material is justified

- by restricting such material to solids, which are in a non-readily dispersible form, therefore explicitly excluding powders as well as liquids or solutions,
- by the need for a leaching test to demonstrate sufficient insolubility of the material when exposed to weather conditions like rainfall (see para. 601.2), and
- by the higher package standard Industrial Package Type 3 (Type IP-3) under non exclusive use conditions which is the same as Type A for solids. In case of Industrial Package Type 2 (Type IP-2), (para. 524), the lack of the water spray test and the penetration test is compensated for by the leaching test and by operational controls under the exclusive use conditions, respectively.

226.10. The specific activity limit for LSA-II liquids of 10^{-5} A_2/g, which is a factor of ten more restrictive than for solids, takes into account that the concentration of a liquid could increase during transport.

226.11. A solid compact binding agent, such as concrete, bitumen, etc., which is mixed with the LSA material, is not considered to be an external shielding material. In this case, the binding agent may decrease the surface radiation level and may be taken into account in determining the average specific activity. However, if radioactive material is surrounded by external shielding material, which itself is not radioactive, as illustrated in Figure 1, this external shielding material is not to be taken into account in determining the specific activity of LSA material.

FIG. 1 Low specific activity material surrounded by a cylindrical volume of non-radioactive shielding material.

226.12. For LSA-II solids, and for LSA-III materials not incorporated in a solid compact binding agent, the Regulations require that the activity be distributed throughout the material. This provision puts no requirement on how the activity is distributed throughout the material, i.e., the activity does not need to be uniformly distributed. It is, however, important to recognize that the concept of limiting the estimated specific activity fails to be meaningful if in a large volume the activity distribution is clearly confined to a small percentage of that volume.

226.13. It is prudent to establish a method by which the significance of the estimated average activity, as determined, can be judged. There are several methods that would be suitable for this particular purpose.

226.14. A simple method for assessing the average activity involves dividing the volume occupied by the LSA material into defined portions and then assessing and comparing the specific activity of each of these portions. It is suggested that specific activity differences between portions of less than a factor of ten would cause no concern.

226.15. Judgement needs to be exercised in selecting the size of the portions to be assessed. The method described in 226.14 should not be used for volumes of material less than 0.2 m^3. For a volume between 0.2 m^3 and 1.0 m^3, the volume should be divided into five, and for a volume greater than 1.0 m^3 into ten parts of approximately equivalent size.

226.16. For LSA-III materials consisting of radioactive material within a solid compact binding agent, the requirement is that they are essentially uniformly distributed in this agent. Because the requirement of `essentially uniformly distributed' for LSA-III materials is qualitative, it is necessary to establish methods by which compliance with the requirement can be judged.

226.17. The following method is an example for LSA-III materials which are essentially uniformly distributed in a solid compact binding agent. The method involves dividing the LSA material volume including the binding agent into a number of portions. At least ten portions should be selected, subject to the volume of each portion being no greater than 0.1 m^3. The specific activity of each volume should then be assessed (either through measurements, calculations, or combinations thereof). It is suggested that specific activity differences between the portions of less than a factor of three would cause no concern. The factor of three in this procedure is more constraining than the suggested factor of ten in para. 226.15 because the requirement that the radioactive material is 'essentially uniformly distributed' is intended to be more constraining than the requirement that the radioactive material is 'distributed throughout'.

226.18. As a consequence of the definition of LSA material additional requirements are specified for:

(a) the quantity of LSA material in a single package with respect to the external radiation level of the unshielded material (see para. 521); and
(b) the total activity of LSA material in any single conveyance (see para. 525 and Table V).

Both requirements can be much more restrictive than the basic requirements for LSA material given in para. 226. This can be seen from the following theoretical example: if it is assumed that a 200 L drum is filled with a solid combustible material having an estimated average specific activity of 2×10^{-3} A_2/g, it would seem that this material could be transported as LSA-III. However, for example, if the density of the material is 1 g/cm^3, the total activity in the drum will be 400 A_2 [$(2 \times 10^{-3}$ A_2/g) (1 g/cm^3) $(2 \times 10^5$ cm^3) = 400 A_2] and transport as LSA-III would be precluded by the conveyance limit of 10 A_2 by inland waterway and by 100 A_2 by other modes (see Table V of the Regulations). See also para. 525.2

226.19. Objects which are both activated or otherwise radioactive and contaminated cannot be considered as SCO (see para. 241.5). However, such objects may qualify as LSA material since an object having activity throughout and also activity distributed on its surfaces may be regarded as complying with the requirement that the activity be distributed throughout. For such objects to qualify as LSA material it is necessary to ascertain that the applicable limits on estimated average specific activity are complied with. In assessing the average specific activity, all radioactive material attributed to the object, i.e., both the distributed activity and the activity of the surface contaminations, needs to be included. As appropriate, additional requirements applicable to LSA material need to also be satisfied .

226.20. Compaction of material should not change the classification of the material. To ensure this, the mass of any container compacted with the material, should not be taken into account when determining the average specific activity of the compacted material.

226.21. See also Appendix I.

Low toxicity alpha emitters

227.1. The identification of low toxicity alpha emitters is based on the specific activity of the radionuclide (or the radionuclide in its as-shipped state). For a nuclide with a very low specific activity, its intake cannot, because of its bulk, reasonably give rise to doses approaching the dose limit. The nuclides uranium-235, uranium-238 and thorium-232 have specific activities 4 to 8 orders of magnitude lower than plutonium-238 or plutonium-239 (4×10^3 to 8×10^4 Bq/g as compared to 2×10^9 to 6×10^{11} Bq/g). Although thorium-228 and thorium-230 have specific activities comparable to those of plutonium-238 and plutonium-239, they are only allowed as 'low toxicity alpha emitters' when contained in ores and physical and chemical concentrates, which inherently provides for the low activity concentration required.

Maximum normal operating pressure

228.1. The maximum normal operating pressure (MNOP) is the difference between the containment system maximum internal pressure and the mean sea-level atmospheric pressure for the conditions specified below.

228.2. The environmental conditions to be applied to a package in determining the MNOP are the normal environmental conditions specified in paras 653 and 654, or in the case of air transport, para. 618. Other conditions to be applied in determining the MNOP are that the package is unattended for a one-year period and that it is subject to the maximum internal heating.

228.3. The one year period exceeds the expected transit time for a package containing radioactive material; besides providing a substantial margin of safety in relation to routine conditions of transport, it also addresses the possibility of loss of a package in transit. The one year period is arbitrary but has been agreed upon as a reasonable upper limit for a package to be given up as lost or remain unaccounted for in transit. Because the package is assumed to be unattended for one year, any physical or chemical changes to the packaging or its contents which are transient in nature and could contribute to increasing pressure of the containment system need to be taken into account. The transient conditions that should be considered include: changes in heat dissipation capability, gas buildup due to radiolysis, corrosion, chemical reactions or release of gas from fuel pins or other encapsulation into the containment system. Some transient conditions may tend to reduce the MNOP such as the reduction in pressure with time caused by a decrease in internal heat due to radioactive decay of the contents. These conditions may be taken into account if adequately justified.

Overpack

229.1. By packing various packages or a single package, each of which fully complies with the requirements of the Regulations, into one overpack the carriage of a consignment from one consignor to one consignee may be facilitated. Specific design, test or approval requirements for the overpack are not necessary since it is the packaging, not the overpack, which performs the protective function. however, the interaction between the overpack and the packages should be taken into account especially concerning the thermal behaviour of the packages during routine and normal conditions of transport.

229.2. A rigid enclosure or consolidation of packages for ease of handling in such a way that package labels remain visible for all packages, need not be considered as an overpack unless advantage is taken by the consignor of the determination of the transport index of the overpack by the direct measurement of the radiation level.

Package

230.1. The terms package and packaging are used to distinguish the assembly of components for containing the radioactive material (packaging), and this assembly of components plus the radioactive contents (package).

230.2. A package is the packaging and its radioactive contents as presented for transport. For design and compliance assurance purposes, this may include any or all structural equipment required for handling or securing the package which is either permanently attached or assembled with the package.

230.3. In order to determine which structural components should be considered part of the package, it is necessary to examine the use and purpose of such equipment with respect to transport. If a package can only be transported with certain structural equipment then it is normal to consider that equipment as part of the packaging. This does not mean that a trailer or transport vehicle should be considered as part of the package in the case of dedicated transport.

230.4. If the package may be transported either with or without certain structural equipment, it may be necessary that both situations be evaluated in determining packaging suitability and compliance.

230.5. If certain equipment is attached during transport for handling purposes, it also may be necessary to consider its effect for normal conditions of transport. In the case of Type B(U), Type B(M), Type C, and packages designed to carry fissile material, the designer must reach agreement with the competent authority for certification. For most other types of packages there is normally no need for specially designed handling or structural equipment, except in large Type IP packages.

230.6. A tank, freight container or intermediate bulk container with radioactive contents may be used as one of the types of package under these regulations provided that it meets the prescribed design, test and any applicable approval requirements for that type of package. Alternatively, a tank, freight container or metal intermediate bulk container with radioactive contents may be used as an industrial package Type IP-2 ot Type IP-3 if it meets the Type IP-1 requirements as well as other requirements which are specifically referenced in paras 625 - 628 of the regulations.

Packaging

231.1. See paras 230.1 and 230.2.

Radiation level

233.1. The limiting quantity in radiological protection for the exposure of people is the effective dose. As this is not a directly measurable quantity, operational quantities had to be created which are measurable. These quantities are ‘ambient dose equivalent’ for strongly penetrating radiation and ‘directional dose equivalent’ for weakly penetrating radiation. The radiation level should be taken as the value of the operational quantitity 'ambient dose equivalent', or 'directional dose equivalent'

233.2. In some cases consideration should be given to the possibility of radiation increase as a result of the buildup of daughter nuclides during transport. In such cases a correction should be applied to represent the highest radiation level envisaged during the transport.

233.3. In mixed gamma and neutron fields it may be necessary to make separate measurements. It should be ensured that the monitoring instrument being used is appropriate for the energy level being emitted by the radionuclide and that the calibration of the instrument is still valid. In performing both the initial measurement and any check measurement, the calibration uncertainties have to be taken into account.

233.4. Neutron dosimeters very often have a significant dependence of the reading with neutron energy. The spectral distribution of the neutrons used for calibration and the spectral distribution of the neutrons to be measured may influence the accuracy of the dose determination considerably. If the energy dependence of the instrument reading and the spectral distribution of the neutrons to be measured are known a corresponding correction factor may be used.

233.5. The Regulations require that, at the surfaces of packages and overpacks, specific radiation levels shall not be exceeded. In most cases a measurement made with a hand instrument held against the surface of the package indicates the reading at some distance away because of the physical size of the detector volume. The instrument used for the measurement of the radiation level should, where practicable, be small in relation to the dimensions of the package or overpack. Large instruments should not be used because they might underestimate the radiation level. Where the distance from the source to the instrument is large in relation to the size of the detector volume (e.g., a factor of 5), the effect is negligible and can be ignored, otherwise the values in Table I should be used to correct the measurement. For radiographic devices where the source to surface distance is generally kept to a minimum, the effect is usually not negligible and an allowance should be made for the size of the detector volume.

TABLE I. CORRECTION FACTORS FOR PACKAGE AND DETECTOR SIZES

Distance between detector centre and package surface (cm)	Half linear dimension of package (cm)	Correction factor [a]
1	> 10	1.0
2	10-20	1.4
	> 20	1.0
5	10-20	2.3
	20-50	1.6
	>50	1.0
10	10-20	4.0
	20-50	2.3
	50-100	1.4
	>100	1.0

[a] The reading should be multiplied by the correction factor to get the actual radiation level at the surface of the packages.

233.6. When monitoring finned flasks or other transport packages, care should be taken where narrow radiation beams may be encountered. A dose rate meter, with a detector area which is much larger than the cross-sectional area of the beam to be measured, will yield a proportionally reduced reading of dose rate because of averaging over the much larger detector area. An appropriate instrument should be chosen for the work involved.

Radioactive material

236.1. In previous editions of the regulations, a single exemption value of 70 Bq/g was used to define radioactive material for transport purposes. Following publication of the Basic Safety Standards[2], it was recognized that this value had no technical basis based upon radiological criteria. Taking into account the radiological protection criteria defined in the BSS, it was established that the BSS exemption values were suitable for transport purposes, (see para. 401.3).

Shipment

237.1. In the context of the transport of radioactive material, the term `destination' means the end point of a journey at which the package is, or is likely to be, opened, except during customs operations as described in para. 581.

Special Arrangement

238.1. The use of the "special arrangement" should not be taken lightly. This type of shipment is intended for those situations where the normal requirements of the regulations cannot be met. For example the disposal of old equipment containing radioactive material where there is no reasonable way to ship the radioactive material in an approved package. The hazard associated with repackaging and handling the radioactive material could outweigh the advantage of using an approved package, assuming a suitable package is available. The special arrangement provisions should compensate for not meeting all the normal requirements of the regulations by providing an equivalent level of safety. In keeping with the underlying philosophy of the transport regulations reliance on administrative measures should be minimized in establishing the compensating measures.

Special form radioactive material

239.1. The Regulations are based on the premise that the potential hazard associated with the transport of non-fissile radioactive material is dependent on four important parameters:

- the dose per unit intake of the radionuclide;
- the total activity contained within the package;
- the physical form of the radionuclide;
- the potential external radiation levels.

239.2. The Regulations acknowledge that radioactive material in an indispersible form or sealed in a strong metallic capsule presents a minimal contamination hazard, although the direct radiation hazard still exists. Material protected in this way from the risk of dispersion during accident conditions is designated as 'special form radioactive material'. Radioactive material which itself is dispersible may be adsorbed, absorbed or bonded to an inert solid in such a manner that it acts as an indispersible solid, e.g., metal foils. See paras 603.1 to 603.4 and paras 604.1 and 604.2.

239.3. Unless the radioactive contents of a package are in special form, the quantity of radioactive material that can be carried in the packaging will be limited to A_2 or multiples thereof. For example, a Type A package is limited to A_2 and the contents of excepted packages are limited to values ranging from A_2 to as low as 10^{-4} A_2, or 10^{-5} A_2 if transported by post, depending upon whether the material is solid, liquid or gas and whether or not it is incorporated into an instrument or article. However, if the material is in special form, the package limits change from A_2 to A_1 or appropriate multiples thereof. Depending on the isotope(s) involved, the A_1 values differ from the A_2 values by factors ranging from 1 to 1000 (see Table I of the Regulations). The capability to ship an increased quantity in a package if it is in special form only applies to Type A and excepted packages.

Specific activity

240.1. The definition of specific activity in practice covers two different situations. The first, the definition of the specific activity of a radionuclide, is similar to the ICRU definition of specific activity of an element. The second, the definition of the specific activity of a material for the Regulations, is more precisely a mass activity concentration. It must be noted that the term “activity concentration” is also used in some paragraphs of the Regulations.

240.2. The half-life and the specific activity for each individual radionuclide given in Table I of the Regulations are shown in Table AII-1 of Appendix II. These values of specific activity were calculated using the following equation:

$$Specific\ activity\ (Bq/g) = \frac{(Avogadro's\ number)\ x\ \lambda}{(Atomic\ mass)}$$

$$= \frac{4.18\ x\ 10^{23}}{A\ x\ T_{1/2}}$$

where A is the atomic mass of the radioisotope.

$T_{1/2}$ is the half-life (in seconds) of the radioisotope
? is the decay constant (seconds^{-1}) of the radioisotope = ln $2/T_{1/2}$.

240.3. The specific activity of any radioisotope not listed in Table AII-1 of Appendix II can be calculated using the equation shown in para. 240.2.

240.4. The specific activity of uranium, for various levels of enrichment, is shown in Table AII-3 of Appendix II.

240.5. In determining the specific activity of a material in which radionuclides are distributed, the entire mass of that material or a subset thereof, i.e., the mass of radionuclides and the mass of any other material, needs to be included in the mass component. The different interpretations of specific activity in the definition of LSA material (para. 226) and in Table AII-1 should be noted.

Surface contaminated object

241.1. A differentiation is made between two categories of surface contaminated objects, in terms of their contamination level, and this defines the type of packaging to be used to transport these objects. The Regulations provide adequate flexibility for the unpackaged shipment of SCO-I objects or the shipment of them in industrial package (Type IP-1). The higher level of non-fixed contamination permitted on objects classified as SCO-II requires the higher standard of containment afforded by industrial package Type IP-2.

241.2. The SCO-I model used as justification for the limits for fixed and non-fixed contamination is based on the following scenario. Objects in the category of surface contaminated objects include those parts of nuclear reactors or other fuel cycle equipment which have come in contact with primary or secondary coolant or process waste resulting in contamination of their surface with mixed fission products. Based on the allowable contamination levels for beta and gamma emitters, an object with a surface area of 10 m^2 could have fixed contamination of up to 4 GBq and non-fixed contamination of up to 0.4 MBq. During routine transport this object can be shipped unpackaged under exclusive use, but it is necessary to secure the object (para. 523(a)) to ensure that the SCO-I object or other cargo cannot move in such a way that the fixed contamination is scraped from the surface of the SCO-I object. In an accident, the SCO-I object and other cargo is assumed to move such that 20 % of the surface of the SCO-I object is scraped, and 20 % of the fixed contamination from the scraped surface is freed. In addition all of the non-fixed contamination is considered to be released. The total radioactivity released would thus be 160 MBq of fixed contamination and 0.4 MBq of non-fixed contamination. Using an A_2 value of 0.02 TBq for mixed beta and gamma emitting fission products, the contamination released equates to 8×10^{-3} A_2. It was considered that such an accident would only occur outside so that, consistent with the basic assumption of the Q-system developed for Type A packages (see Appendix I), an intake of 10^{-4} of the scraped radionuclides for a person in the vicinity of the accident is appropriate. This would result in a total intake of 0.8×10^{-6} A_2. Hence this provides a level of safety equivalent to that of Type A packages.

241.3. The model for a SCO-II object is similar to that for a SCO-I object, although there may be up to 20 times as much fixed contamination and 100 times as much non-fixed contamination. However an industrial package (IP-2) is required for the transport of SCO-II objects. The presence of this package will lead to a release fraction in an accident which approaches that for a Type A package. Using a release fraction of 10^{-2} results in a total release of beta and gamma emitting activity of 32 MBq of fixed contamination and 8 MBq of non fixed contamination, which equates to 2×10^{-3} A_2. Applying the same intake factor as in the previous paragraph leads to an intake of 0.2×10^{-6} A_2, thereby providing a level of safety equivalent to that of Type A packages.

241.4. If the total activity of a surface contaminated object is so low, that the activity limits for excepted packages according to para. 408 are met, it can be transported as an excepted package provided that all the applicable requirements and controls for transport of excepted packages (paras 515 - 519) are complied with.

241.5. Surface contaminated objects (SCO) are by definition objects which are themselves not radioactive, but have radioactive materials distributed on their surfaces. The implication of this definition is that objects that are radioactive themselves (e.g., activated objects) and are also contaminated cannot be classified as SCO. Such objects may, however, be regarded as LSA material insofar as the requirements specified in the LSA definition are complied with. See also para. 226.19.

241.6. Examples of inaccessible surfaces are:

- inner surfaces of pipes, the ends of which can be securely closed by simple methods;
- inner surfaces of maintenance equipment for nuclear facilities which are suitably blanked off or formally closed; and
- glove boxes with access ports blanked off.

241.7. Measurement techniques for fixed and non-fixed contamination of packages and conveyances are given in paras 508.2 and 508.7 to 508.12. These techniques are applicable to SCO materials. However, to properly apply these techniques a consignor needs to know the composition of the contamination. (See also para.523.1)

Tank

242.1. The lower capacity limit of 450 litres (1000 litres in the case of gases) is included to achieve harmonization with Chapter 12 of the United Nations Recommendations [7].

242.2. Para. 242 includes solid contents in tanks where such contents are placed in the tank in liquid or gaseous form and subsequently solidified prior to transport (for example uranium hexafluoride, UF_6).

Transport index

243.1. The transport index performs many functions in the Regulations, including providing the basis for the carrier to segregate radioactive materials from persons, undeveloped film, and other radioactive material consignments and to limit the level of radiation exposure to members of the public and transport workers during transport and in-transit storage.

243.2. In the 1996 Edition of the Regulations the TI no longer has any contribution to the criticality safety control of packages containing fissile material. Control for criticality safety is now provided by a separate criticality safety index (CSI) (see paras 218.1 and 218.2). Although the previous approach of a single control value for radiological protection and criticality safety provided for simple operational application, the current use of a separate TI and CSI removes significant limitations on segregation in the transport and storage in transit of packages not containing fissile material. The reason for retaining the designation of “TI” is that the vast majority of radioactive consignments are not carrying fissile material, and that, therefore, a new name for the "radioactive only" TI could have created confusion, because of the need to introduce and explain two new names. Care should be taken not to confuse the use of the TI value and to consider the CSI value as the only control for accumulation of packages for criticality safety.

243.3. See paras 526.1 to 526.4.

Unirradiated thorium

244.1. The term `unirradiated thorium' in the definition of low specific activity material is intended to exclude any thorium which has been irradiated in a nuclear reactor so as to transform some of the thorium-232 into uranium-233, a fissile material. The definition could have prohibited the presence of any uranium-233, but all naturally occurring thorium contains trace amounts of uranium-233. The limit of 10^{-7} g of uranium-233 per gram of thorium-232 is intended to clearly prohibit any irradiated thorium while recognizing the presence of trace amounts of uranium-233 in all natural thorium.

Unirradiated uranium

245.1. The term `unirradiated uranium' is intended to exclude any uranium which has been irradiated in a nuclear reactor so as to transform some of the uranium-238 into plutonium-239 and some of the uranium-235 into fission products. The definition could have prohibited the presence of any plutonium or fission products, but all naturally occurring uranium contains trace amounts of plutonium and fission products. The limits of 10^{-6} g of plutonium per gram of uranium-235 and 9 MBq of fission products per gram of uranium-235 are intended to clearly prohibit any irradiated uranium while recognizing the presence of trace amounts of plutonium and fission products in all natural uranium. The defined terms in paras 245 and 246 have been combined in para. 226(a)(ii) to define acceptable forms of LSA-I.

245.2. The presence of uranium-236 is a more satisfactory indication of exposure to a neutron flux. $5x10^{-3}$ grams of uranium-236 per gram uranium-235 has been chosen as representing the concensus view of ASTM Committee C-26 in specification C-996 for enriched commercial grade uranium. This value recognizes the possibility for trace contamination by irradiated uranium but provides that the material may still be treated as unirradiated. This specification represents the maximum value for uranium isotopes for which composition the A_2 value can be demonstrated to be unlimited for uranium hexafluoride. The difference in A_2 for uranium dioxide is considered to be insignificant [8].

Uranium - natural, depleted, enriched

246.1. See para. 245.1.

SECTION III

GENERAL PROVISIONS

RADIATION PROTECTION

301.1. The objectives of the Radiation Protection Programme (RPP) for the transport of radioactive material is to:

(a) provide for adequate consideration of radiation protection measures in transport;
(b) ensure that the system of radiological protection is adequately applied;
(c) enhance a safety culture in the transport of radioactive material; and
(d) provide practical measures to meet these objectives.

The RPP should include, to the extent appropriate, the following elements:

(a) scope of the programme (see paras 301.2 - 301.4);
(b) roles and responsibilities for the implementation of the programme (see para. 301.5);
(c) dose assessment (see para. 305);
(d) surface contamination assessment (see paras 508, 513 and 514);
(e) dose limits, dose constraints and optimization (see para. 302);
(f) segregation distances (see paras 306 - 307);
(g) emergency response (see paras 308 - 309);
(h) training (see para. 303), and
(i) quality assurance (see para. 310).

301.2. The scope of the Radiation Protection Programme (RPP) should include all the aspects of transport as defined in para. 106 of the Regulations. However, it is recognized that in some cases certain aspects of the RPP may be covered in RPPs at the consigning, receiving or storage-in-transit sites. Since the magnitude and extent of measures to be employed in the RPPs will depend on the magnitude and likelihood of exposures involved, a graded approach should be followed.

301.3. Both the package type and the package category need to be considered. For routine transport the external radiation is important and the package category provides a classification for this; under accident conditions however it is the package type (excepted, industrial, Type A, Type B or Type C) that is important. Excepted, industrial, and Type A packages are not required to withstand accidents. Those aspects of the RPP related to accident conditions of transport will need to consider the possibility of leakage from these package types as the result of relatively minor transport or handling accidents. In contrast Type B and Type C packages can be expected to withstand all but very severe accidents.

301.4. The external radiation levels from excepted packages and Category I white label packages are sufficiently low that they are safe to handle without restriction and a dose assessment (301.3) is therefore unnecessary. Consideration of dose limits, constraints and optimization can be limited to keeping handling times as low as reasonably achievable (301.4), and segregation (301.5) can be met by avoiding prolonged direct contact of packages with persons and other goods during transport. A dose assessment will, however, be needed for Category II and III yellow label packages and segregation, dose limits, constraints, and optimization will need to be considered in its light.

301.5. The radiation protection programme will best be established through the cooperative effort of consignors, carriers and consignees engaged in the transport of radioactive material. Consignors and consignees should normally have an appropriate radiation protection programme as part of fixed facility operations. The role and responsibilities of the different parties and individuals involved in the implementation of the RPP should clearly be identified and described. Overlapping of responsibilities should be avoided. Depending on the magnitude and likelihood of radiation exposures, the overall responsibility for the establishment and the implementation of the RPP may be attributed to a health physics or safety officer recognized by the competent authorities (called "qualified expert" in the European Basic Safety Standards for Radiation Protection).

302.1. Optimization of protection and safety requires that both normal and potential exposures are taken into account. Normal exposures are exposures which are expected to be received under routine and normal transport conditions as defined in para. 106 of the Regulations. Potential exposures are exposures which are not expected to be delivered with certainty but that may result from an accident or owing to an event or sequence of events of a probabilistic nature, including equipment failures and operating errors. In the case of normal exposures, optimization requires that the expected magnitude of individual doses and the number of people exposed are taken into account; in addition, in the case of potential exposures, the likelihood of the occurrence of accidents or events or sequences of events is also taken into account. Optimization has to be documented in the RPPs. See also TECDOC 374 'Discussion of and Guidance on the Optimization of Radiation Protection in the Transport of Radioactive Material' [9].

302.2. The IAEA Basic Safety Standards (SS 115) [2] defines two systems of radiological protection. One system is for practices (activities that increase the overall exposure to radiation) and the other is for intervention (activities that decrease the overall exposure by influencing the existing causes of exposure). The system of radiological protection for practices is summarized as follows :

- No practice shall be adopted unless it produces a positive net benefit (justification of a practice)
- All exposures should be kept as low as reasonably achievable, economic and social factors being taken into account (optimization of protection)
- Total individual exposure should be subject to dose limits or in the case of potential exposures to the control of risk (individual dose and risk limits).

302.3. In practical radiological protection there is a need to provide standards associated with quantities other than the basic dose limits. Standards of this type are normally known as secondary or derived limits. When such limits are related to the primary limits of dose by a defined model, they are referred to as derived limits.

302.4. Examples of derived limits in the Regulations include the maximum activity limits A_1 and A_2, maximum levels for non-fixed contamination, radiation levels at the surfaces of packages and in their proximity, and segregation distances associated with the transport index. The Regulations stress the importance of assessment and measurement to ensure that standards are being complied with.

302.5. It should be a task of the competent authority to ensure that all transport activities are conducted under a general framework of optimization.

303.1. The provision of information and training is an integral part of any system of radiological protection. The level of instruction provided should be appropriate to the nature and type of work undertaken. Workers involved with the transport of radioactive material require training concerning the radiological risks involved in their work and how they can minimize these risks in all circumstances.

303.2. Training should relate to specific jobs and duties, to specific protective measures to be undertaken in the event of an accident, or to the use of specific equipment. It should include general, information relating to the nature of radiological risk, and knowledge of the nature of ionizing radiations, their effects, and their measurement, as appropriate. Training should be seen as a continuous commitment throughout employment and involves initial training and refresher courses at appropriate intervals. The effectiveness of the training should be periodically checked.

303.3. Information on specific training requirements has been published.[10],[11]

304.1. The competent authority assessments may be used to evaluate the effectiveness of the Regulations including those for RPPs and may be part of the compliance assurance activities detailed in IAEA Safety Series No.112 [12] (see also paras 311.1 to 311.8). Of particular importance is the assessment of whether there is effective optimization of radiation protection. This is to ensure doses to workers and members of the public are below regulatory limits and kept as low as reasonably achievable and will help to achieve and maintain public confidence.

304.2. In order to comply with para. 304 of the Regulations, information on the radiation doses to workers and to members of the public should be collected and reviewed as appropriate. Reviews should be made if circumstances warrant, e.g., if significant changes in transport patterns occur or when a new radioactive material-related technology is introduced. The collection of relevant information may be achieved through a combination of radiation measurements and assessments. Reviews of accident conditions of transport are necessary in addition to those of routine and normal conditions.

305.1. The Basic Safety Standards [2] set a limit on effective dose for the members of the public of 1 mSv/year, and for workers 20 mSv/year averaged over five consecutive years and not exceeding 50 mSv in a single year. Dose limits in special circumstances, dose limits in terms of equivalent dose for the lens of the eye, extremities (hand and feet) and skin, and dose limits for apprentices and pregnant women are also set out in Basic Safety Standards and should be considered in the context of the requirements of para. 305. These limits apply to exposures attributable to all practices, with the exception of medical exposures and of exposures from certain natural sources.

305.2. Three categories for monitoring and assessing radiation doses are shown in para. 305. The first category establishes a dose range where little action needs to be taken for evaluating and controlling doses. The upper value of this range was chosen to coincide with the dose limit for a member of the public. The second category has a value of 6 mSv/year which is roughly 1/3 of the limit on effective dose for workers and represents a reasonable dividing line between conditions where dose limits could be approached and conditions where they are unlikely to be approached.

305.3. Many transport workers will be in the first category and no specific measures concerning monitoring or control of exposure are required. In the second category, a dose assessment programme will be necessary. This may be based upon either individual monitoring or monitoring of the workplace. In the latter case, workplace monitoring may often be satisfied by radiation level measurements in occupied areas at the start and end of a particular stage of a journey. In some cases, however, air monitoring, surface contamination checks and individual monitoring may also be required. In the third category individual monitoring is mandatory. In most cases this will be accomplished by the use of personal dosimetry such as film badges, TLD's and, where necessary, neutron dosimeters.

305.4. Some studies of particular operations have shown a correlation between dose received by workers and the transport index handled.[13] It is unlikely that carriers handling less than 300 TI per year will exceed doses of 1 mSv/year and such carriers would not therefore require detailed monitoring, dose assessment, or individual records.

305.5. Taking into account that relatively high radiation levels are permitted during carriage under exclusive use additional care must be taken to ensure that the requirements of para. 305 are met, since it would be relatively easy to exceed the 1 mSv level and consequently specific measures regarding monitoring or control of exposures are needed. In the assessment of the correct exposure category it will be necessary to consider exposures received during the carriage phase of transport together with those received elsewhere particularly during loading and unloading.

306.1. The dose level of 5 mSv per year for occupationally exposed workers and 1 mSv per year to the critical group [2] for members of the public are specifically defined values to be used for the purposes of calculating segregation distances or dose rates in regularly occupied areas. The distances and dose rates are, for convenience, often presented in segregation tables. These values are for segregation distance or calculation purposes only and are required to be used together with hypothetical but realistic parameters in order to obtain appropriate segregation distances. Using the given values provides reasonable assurance that actual doses from the transport of radioactive materials will be well below the appropriate average annual dose limits.

306.2. These values together with simple, robust modelling have been used for a number of years to derive segregation tables for different modes of transport. Assessments of radiation exposures arising indicate that continued use of these values is acceptable. In particular, surveys of exposure occurring in air and sea transport [14],[15] have shown that segregation distances derived from them have resulted in doses to the public below the relevant annual dose limits and that doses to workers not involved in direct handling are also less than 1 mSv. The use of segregation distances does not in itself however take away the requirement from undertaking the evaluation required in para. 305 of the Regulations.

306.3. The Regulations state the principles of radiation protection which are to be applied in the determination through calculations of segregation distances (i.e., minimum distances between radioactive material packages and regularly occupied areas of a conveyance) and of dose rates in regularly occupied areas. For practical purposes it may be helpful to provide this information in the form of segregation tables.

307.1. An evaluation of the effect of radiation on fast X-ray films in 1947 [16] determined that they may show slight fogging after development when exposed to doses exceeding 0.15 mSv of gamma radiation. This could interfere with the proper use of the film and provide incorrect diagnostic interpretation. Other types of film are also susceptible to fogging although the doses required are much higher. Since it would be impracticable to introduce segregation procedures which varied with the type of film, the provisions of the regulations are designed to restrict the exposure of undeveloped films of all kinds to a level of not more than 0.1 mSv during any journey from consignor to consignee.

307.2. The different time durations involved for sea transport (in terms of days or weeks) and air or land transport (in terms of hours or days), mean that different tables of segregation distances are used so that the total film exposure during transit is the same for each mode. More than one mode of transport and more than one shipment may be involved in the distribution and ultimate use of photographic film. Thus, when segregation distance tables are being established for a specific transport mode, only a fraction of the limit prescribed in para. 307 should be committed to that mode.

307.3. In road transport a driver may ensure sufficient segregation from photographic film carried in other vehicles by leaving a clear space of at least 2 m all round the vehicle when parking.

EMERGENCY RESPONSE

308.1. The standards prescribed by the Regulations, when complied with by the package designer, consignor, carrier and consignee, ensure a very high level of safety for the transport of radioactive material. However, accidents involving such packages may happen. Para. 308 of the Regulations recognizes that advance planning and preparation are required to provide a sufficient and safe response to such accidents. The response, in most cases, will be similar to the response to radiation accidents at fixed site facilities. Thus, it is required that relevant national or international organizations establish emergency procedures, and that in the event of a transport accident involving radioactive materials, these procedures are followed.

308.2. Further guidance can be found in IAEA Safety Series No. 87 [17].

309.1. The radioactive hazard may not be the only potential hazard posed by the contents of a package of radioactive material. Other hazards may exist, including pyrophoricity, corrosivity or oxidizing properties; or, if released, the contents may react with the environment (air, water, etc.), in turn producing hazardous substances. It is this latter phenomenon which para. 309 of the Regulations addresses so as to ensure proper safety from chemical (i.e., non-radioactive) hazards, and specific attention is drawn to uranium hexafluoride (UF_6) because of its propensity to react, under certain conditions, both with humidity in the air and with water to form hydrogen fluoride and uranyl fluoride (HF and UO_2F_2).

309.2. In the event that the containment system of a package is damaged in an accident, air and/or water may reach and, in some cases, chemically react with the contents. For some radioactive materials, these chemical reactions may produce caustic, acidic, toxic or poisonous substances which could be dangerous to people and the environment. Consideration should be given to this problem in the design of the package and in emergency response planning procedures to reduce the consequences of such reactions. In doing so, the quantities of materials involved, the potential reaction kinetics, the ameliorating effects of reaction products (self- extinguishing, self-plugging, insolubility, etc.), and the potential for concentration or dilution within the environment should all be considered. Such considerations may lead to restrictions on the package design, or its use, which go beyond considerations of the radioactive nature of the contents.

QUALITY ASSURANCE

310.1. Quality assurance is essentially a systematic and documented method to ensure that the required conditions or levels of safety are consistently achieved. Any systematic evaluation and documentation of performance judged against an appropriate standard is a form of quality assurance. A disciplined approach to all activities affecting quality, including, where appropriate, specification and verification of satisfactory performance and/or implementation of appropriate corrective actions, will contribute to transport safety and provide evidence that the required quality has been achieved.

310.2. The Regulations do not prescribe detailed quality assurance programmes because of the wide diversity of operational needs and the somewhat differing requirements of the competent authorities of each Member State. A framework upon which all quality assurance programmes may be based is provided in Appendix IV. The degree of detail in the quality assurance programme will depend on the phase and type of transport operation, adopting a graded approach consistent with paragraph 104 of the Regulations.

310.3 The development and application of quality assurance programmes, as required by the Regulations, should be carried out in a timely manner, before transport operations commence. Where appropriate, the Competent Authority will ensure that such quality assurance programmes are implemented, as part of the timely adoption of the Regulations.

310.4. Further guidance is given in IAEA Safety Series No. 113 [18].

COMPLIANCE ASSURANCE

311.1. The adoption of transport safety regulations, based on the Regulations, is carried out within an appropriate time frame in Member States and by all relevant international organizations. Emphasis is placed on the timely implementation of systematic compliance assurance programmes to complement the adoption of the Regulations.

311.2. As used in the Regulations, the term `compliance assurance' has a very broad meaning which includes all of the measures applied by a competent authority which are intended to ensure that the provisions of the Regulations are complied with in practice.
Compliance means, for example, that:

(a) Appropriate and sound packages are used;
(b) The activity of radioactive material in each package does not exceed the regulatory activity limit for that material and that package type;
(c) The radiation levels external to, and the contamination levels on, surfaces of packages do not exceed the appropriate limits;
(d) Packages are properly marked and labelled and transport documents are completed;
(e) The number of packages containing radioactive material in a conveyance is within the regulatory limits;
(f) Packages of radioactive material are stowed in conveyances and are stored at a safe distance from persons and photosensitive materials;
(g) Only those stowage and lifting devices which have been tested are used in loading, conveying and unloading packages of radioactive material (see para. 564); and
(h) Packages of radioactive material are properly secured for transport.
(i) Only trained personnel are used for handling radioactive material during transport operations. This includes drivers of vehicles carrying radioactive material.

311.3. The principal objectives of a systematic programme of compliance assurance are:

- To provide independent verification of regulatory compliance by the users of the Regulations;
- To provide feedback to the regulatory process as a basis for improvements to the Regulations and the compliance assurance programme.

311.4. An effective compliance assurance programme should as a minimum include measures related to:

- Review and assessment, including the issuance of approval certificates,
- Inspection and enforcement.

311.5. A compliance assurance programme can only be implemented if its scope and objectives are conveyed to all parties involved in the transport of radioactive materials, i.e., designers, manufacturers, consignors and carriers. Therefore compliance assurance programmes should include provisions for information dissemination. This should inform users about the way the competent authority expects them to comply with the Regulations and about new developments in the regulatory field. All parties involved should have trained staff.

311.6. In order to ensure the adequacy of special form radioactive material (see para. 239 of the Regulations) and certain package designs, the competent authority is required to assess these designs (see para. 802 of the Regulations). In this way the competent authority can ensure that the designs meet the regulatory requirements and that the requirements are applied in a consistent manner by different users. When required by the Regulations, shipments are also subject to review and approval in order to ensure that adequate safety arrangements are made for transport operations.

311.7. It is necessary for the competent authority to perform audits and inspections as part of its compliance assurance programme in order to confirm that the users are meeting all applicable requirements of the Regulations and are applying their quality assurance programmes. Inspections are also necessary to identify instances of non-compliance which may need either corrective action by the user or enforcement action by the competent authority. The primary purpose of an enforcement programme is not to carry out punitive action, but to foster compliance with the Regulations.

311.8. Since the Regulations include requirements for emergency provisions during the transport of radioactive materials (see para. 308 of the Regulations), a compliance assurance programme should include activities pertaining to emergency planning and preparedness and to emergency response when needed. These activities should be incorporated into the appropriate national emergency plans. The appropriate competent authority should also ensure that consignors and carriers have adequate emergency plans.

311.9. Further guidance is given in IAEA Safety Series No. 112 [12].

SPECIAL ARRANGEMENT

312.1. The intent of para. 312 of the Regulations is consistent with similar provisions in the earlier editions of the Regulations. Indeed, the Regulations have, from the earliest edition in 1961, permitted the transport of consignments not satisfying all the specifically applicable requirements, but this can only be done under special arrangement. Special arrangement is based on the requirement that the overall level of safety resulting from additional operational control must be shown to be at least equivalent to that which would be provided had all applicable provisions been met (see para. 104.1). Because the normally applicable regulatory requirements are not being satisfied, each special arrangement must be specifically approved by all competent authorities involved (i.e., multilateral approval is required).

312.2. The concept of special arrangement is intended to give flexibility to consignors to propose alternative safety measures effectively equivalent to those prescribed in the Regulations. This makes possible both the development of new controls and techniques to satisfy the existing and changing needs of industry in a longer term sense and the use of special operational measures for particular consignments where there may be only a short term interest. Indeed, the role of special arrangement as a possible means of introducing and testing new safety techniques which can later be assimilated into specific regulatory provisions is also vital as regards the further development of the Regulations.

312.3. It is recognized that unplanned situations may arise during transport, such as a package suffering minor damage or in some way not meeting all the relevant requirements of the Regulations, which will require action to be taken. When there is no immediate health, safety or physical security concern, a special arrangement may be appropriate. Special arrangements should not be required to deal with occurrences of non-compliance which may require immediate transport to bring the non-compliant situation under appropriate health and safety controls. It is considered that the emergency response procedures of IAEA Safety Series No. 87 [17] and the compliance assurance programmes of IAEA Safety Series No. 112 [12] provide better approaches in most cases for these types of unplanned events.

312.4. Approval under special arrangement can be sought in respect of shipments where variations from standard package design features result in the need to apply compensatory safety measures in the form of more stringent operational controls. Details of possible additional controls which can be used in practice for this purpose are included in para. 825. Information supplied to support equivalent safety arguments may comprise quantitative data, where available, and may range from considered judgment based on relevant experience to probabilistic risk analysis.

SECTION IV

ACTIVITY LIMITS AND MATERIAL RESTRICTIONS

BASIC RADIONUCLIDE VALUES

401.1. The activity limitation on the contents of Type A packages (A_1 for special form material and A_2 for material not in special form) for any radionuclide or combination of radionuclides is derived on the basis of the radiological consequences which are deemed to be acceptable, within the principles of radiological protection, following failure of the package after an accident. The method of deriving A_1 and A_2 values is given in Appendix I.

401.2. The Regulations do not prescribe restrictions on the number of Type A packages transported on a conveyance. It is not unusual for Type A packages to be transported together, sometimes in large numbers. As a result, it is possible for the source term in the event of an accident involving these shipments to be greater than the release from a single damaged package. However, it is considered unnecessary to constrain the size of the potential source term by limiting the number of Type A packages on a conveyance. Most Type A packages carry a small fraction of an A_1 or A_2 quantity, indeed only a small percentage of consignments of Type A packages comprise more than the equivalent of one full Type A package. A study undertaken in the UK [19] found that the highest loading of a conveyance with many Type A packages was equivalent to less than five full Type A packages. Experience also indicates that Type A packages perform well in many accident conditions. Combining event data from the USA [20] and UK [21] over a period of about twenty years provides information on twenty-two accidents involving consignments of multiple Type A packages. There was a release of radioactive contents in only two of these events. Both led to releases in the order of 10^{-4} A_2. A further example can be found in the description of an accident that happened in the USA in 1983 [22] where a conveyance carrying 82 packages (Type A and excepted) with a total approximately 4 A_2 on board. Two packages were destroyed releasing approximately 10^{-4} A_2 of activity.

401.3. The general principles for exemption [2] are that:

(a) the radiation risks to individuals caused by the exempted practice or source be sufficiently low as to be of no regulatory concern;
(b) the collective radiological impact of the exempted practice or source be sufficiently low as not to warrant regulatory control under the prevailing circumstances; and
(c) the exempted practices and sources be inherently safe, with no appreciable likelihood of scenarios that could lead to a failure to meet the criteria in (a) and (b).

Exemption values in terms of activity concentrations and total activity have been derived for inclusion in the Basic Safety Standards on the following basis [23]:

a) an individual effective dose of 10 µSv/y for normal conditions;
b) a collective dose of 1 man Sv per year of practice for normal conditions;
c) an individual effective dose of 1 mSv for accidental conditions;
d) an individual dose to the skin of 50 mSv for both normal and accidental conditions.

These levels were derived on the basis of a variety of exposure scenarios and pathways that did not explicitly address the transport of radioactive material. Additional calculations were performed for transport-specific scenarios [24]. These transport specific exemption values were then compared with the values in the Basic Safety Standards. It was concluded that the relatively small differences between both sets did not justify the incorporation in the Regulations of a set of exemption values different from the one in the Basic Safety Standards, recognizing that the use of different exemption values in various practices may give rise to problems at interfaces and may cause legal and procedural complications.

For radionuclides not listed in the Basic Safety Standards, exemption values have been calculated using the same methodology.

401.4. The activity concentration values are to be applied to the radioactive material within a packaging or a conveyance.

401.5. Exemption values for "total activity" have been established for the transport of small quantities of material for which the total activity, when transported together, is unlikely to result in any significant radiological exposure even when exemption values for "activity concentration" are exceeded. The exemption values for "total activity" are therefore established on a per consignment basis rather than on a per package basis.

401.6. It must be emphasized that, in the case of decay chains, the values in Table I columns 4 and 5 of the Regulations relate to the activity or activity concentration of the parent nuclide.

DETERMINATION OF BASIC RADIONUCLIDE VALUES

403.1. In the event that A_1 or A_2 values need to be calculated the methods outlined in Appendix I should be used. Two situations are considered here. First, a decay chain including one or more radionuclides in equilibrium in which the half lives of all daughters are less than ten days and in which no daughter has a half-life more than the parent nuclide, and second, any other situation. In the former case only the chain parent need be considered because the contribution of the daughters was considered in developing the A_1/A_2 values (see Appendix I); whereas, in the latter case, all the nuclides should be considered separately and considered as a mixture of radionuclides in accordance with para. 404 of the Regulations.

403.2 In the event that exemption values need to be calculated, the methods used to derive the values in the Basic Safety Standards, as outlined in European Commission Radiation Protection Report No. 65 [23], should be used.

404.1. See Appendix I.

404.2. Reactor plutonium recovered from low enriched uranium spent fuel (less than 5% uranium-235) constitutes a typical example of a mixture of radionuclides with known identity and quantity for each constituent. Calculations according to para. 404 of the Regulations result in activity limits independent of the abundance of the plutonium isotopes and the burnup within the range 10 000 MW d/t to 40 000 MW d/t. The following values for reactor plutonium can be used within the above range of burnup taking into account the Am-241 buildup, up to five years after recovery:

$A_1 = 20$ TBq
$A_2 = 3 \times 10^{-3}$ TBq

It is emphasized that these values can be applied only to the case of plutonium separated from spent fuel from thermal reactors, where the original fuel comprised uranium enriched up to 5% in uranium-235, where the burnup was in the range not less than 10 000 MW d/t to not more than 40 000

MW d/t, and where the separation was carried out less than five years before completion of the transport operation. It will also be necessary to separately consider other contaminants in the plutonium.

405.1. In the case of mixtures of radionuclides where the identity is known but the relative proportions are not known in detail, a simplified method to determine the basic radionuclide values is given. This is particularly useful in the case of mixed fission products, which will almost invariably contain a proportion of transuranic nuclides. In this case the grouping would simply be between alpha emitters and other emitters, using the most restrictive of the respective basic radionuclide values for the individual nuclides within each of the two groups. Knowledge of the total alpha activity and remaining activity is necessary to determine the activity limits on the contents. Using this method for the particular fission product mixture present, it is possible to account for both the risk from transuranic elements and that from the fission products themselves. The relative risks will depend upon the origin of the mixture, i.e., the fissionable nuclide origin, the irradiation time, the decay time and possibly the effects of chemical processing.

405.2. For reprocessed uranium, A_2 values may be calculated by using the equation for mixtures in para. 404 taking account of the physical and chemical characteristics likely to arise in both normal and accident conditions. It may also be possible to demonstrate that the A_2 value is unlimited by showing that 10 mg of the uranium contains less activity than that giving rise to a committed effective dose of 50 mSv for that mixture. In addition, for calculating A_2 values in the case of reprocessed uranium, the advice provided in IAEA TECDOC-750 "Interim guidance for the transport of reprocessed uranium" [8] may provide useful information.

405.3. In the case of mixtures of radionuclides whose individual identities are known but individual activities are not known in detail it may be possible to allocate nuclides to groups, taking the lowest basic radionuclide value, as appropriate, for each group to apply to all radionuclides within the group. Thus, if the total activity in each group is known and the lowest value for a member of that group is known, a composite value for the mixture can be determined. The composite value will be lower than the highest group value to allow for the contribution from other groups. This method is most appropriate if mixtures of alpha and beta/gamma radionuclides are present (e.g., mixed fission products associated with transuranics). In this case knowledge of the total alpha activity and total beta/gamma activity is required together with knowledge of the most restrictive basic radionuclide values for the alpha emitters and beta/gamma emitters present.

406.1. Table II of the Regulations provides default data for use in the absence of known data. The values are the lowest possible values within the alpha or beta/gamma subgroups.

CONTENTS LIMITS FOR PACKAGES

Excepted packages

409.1. Articles manufactured of natural or depleted uranium are by definition LSA-I and hence would normally have to be transported in an industrial package Type 1 (IP-1). However, provided the materials are contained in a sheath to prevent oxidation or abrasion they may be transported in excepted packages. The sheath would also absorb all alpha radiation, reduce the beta radiation levels and reduce the potential risk of internal contamination.

410.1. See para. 579.1.

Industrial packages Type 1, Type 2 and Type 3

411.1. See paras 521.1 and 525.1.

Type B(U) and Type B(M) packages

415.1. Content limits for Type B(U) and Type B(M) packages are specified on the approval certificates.

416.1. For Type B(U) and Type B(M) packages to be transported by air the content limits are further restricted to the lower of 3000 A_1 or 100 000 A_2 for special form material and 3000 A_2 for all other radioactive material.

416.2 The 3000 A_2 limit for non-special form material was established taking into account risk analysis work by Hubert et al.[25] concerning Type B package performance in air transport accidents. It is also the threshold quantity for which shipment approval of Type B(M) packages is required.

416.3. With regard to the radioactive content limit for special form radioactive material, it follows from the Q system that 3000 A_1 was adopted as the radioactive content limit for such material in parallel to the 3000 A_2 radioactive content limit. However, for certain alpha emitters the ratio A_1 to A_2 can be as high as 10^4, which would lead to effective potential package loadings of $3x10^7$ A_2 not in dispersible form. This was seen as an undesirably high level of radioactive content particularly if the special form was partially disrupted in a very severe accident. It was assumed that the similarity between the special form impact test and the Type B impact test, implies that special form may be expected to provide a comparable 10^2 reduction in release as indicated above for Type B package allowing the source to increase by a factor of 100 to 300 000 A_2. The value of 100 000 A_2 was taken as a conservative estimate.

416.4. Radioactive material in a non-dispersible form or sealed in a strong metallic capsule presents a minimal contamination hazard, although the direct radiation hazard still exists. Additional protection provided by the special form definition is sufficient to ship special form material by air in a Type B(U) package up to an activity of 3000 A_1 but not more than 100 000 A_2 of the special form nuclide. French studies indicated that some special form material approved under current standards may retain its containment function under test conditions for air accidents [25].

Type C packages

417.1. The design of a Type C package must limit the potential releases to acceptable levels should the package be involved in a severe air accident. The content limits for Type C packages, as specified on the approval certificates, take into account the testing requirements for a Type C package which reflect the potentially very severe accident forces which could be encountered in a severe air transport accident. The design must also ensure that the form of the material and the physical or chemical states are compatible with the containment system.

Packages containing fissile material

418.1. It is important that the fissile material and any other contents material relied on for criticality safety should comply with the approved description of the package contents because criticality safety can be sensitive to the quantity, type, form and configuration of fissile material, any fixed neutron poisons, and/or other non-fissile material included in the contents. Care should be taken to include in the description of the authorized contents any materials (e.g., inner receptacles, packing materials, void displacement pieces, etc.) or significant impurities that possibly or inherently may be present in the package. Thus, the safety assessment should carefully consider the full range of parameters that characterize all material intended as possible contents in the package. Compliance with the quantity of fissile material specified in the certificate of approval is important because any change could cause a higher neutron multiplication factor due to more fissile material or, in the case of less fissile material, could potentially allow a higher reactivity caused by an altered optimal water moderation (for example the certificate may need to insist on complete fuel assemblies being shipped - with no pins removed). Including fissile material or other radionuclides not authorized for the package can have an unexpected effect on criticality safety (for example, replacing ^{235}U by ^{233}U can yield a higher multiplication factor). Similarly, the placement of the same quantity of fissile material in a heterogeneous or homogeneous distribution can significantly affect the multiplication factor. A heterogeneous lattice arrangement provides a higher reactivity for low-enriched uranium systems than a homogeneous distribution of the same quantity of material.

Packages containing uranium hexafluoride

419.1. The limit for the mass of uranium hexafluoride in a loaded package is specified in order to prevent over-pressurization both during filling and emptying. This limit should be based upon the maximum uranium hexafluoride working temperature of the cylinder, the certified minimum internal volume of the cylinder, a minimum uranium hexafluoride purity of 99.5%, and a minimum safety margin of 5% free volume when the uranium hexafluoride is in the liquid state at the maximum working temperature (see ISO 7195 [26]).

419.2. The requirement that the uranium hexafluoride be in solid form and that the internal pressure inside the uranium hexafluoride cylinder be below atmospheric pressure when presented for transport was established as a safe method of operation and to provide the maximum possible safety margin for transport. Generally, cylinders are filled with uranium hexafluoride at pressures above atmospheric pressure under gaseous or liquid conditions. Until the uranium hexafluoride is cooled and solidified, a failure of the containment system in either the cylinder or the associated plant fill system could result in a dangerous release of uranium hexafluoride. However, since the triple point of uranium hexafluoride is $64^{o}C$ at normal atmospheric pressure of 1.013×10^{5} Pa, if the uranium hexafluoride is presented for transport in a thermally steady-state, solid condition, it is unlikely that during normal conditions of transport it will exceed the triple point temperature.

419.3. Satisfying the requirement that the uranium hexafluoride be in solid form with an internal cylinder pressure less than atmospheric for transport ensures that: (a) the handling of the cylinder prior to and following transport and the transport under normal conditions will occur with the greatest safety margin relative to the package performance; (b) the structural capabilities of the package are maximized; and (c) the containment boundary of the package is functioning properly. Satisfying this requirement precludes cylinders being presented for transport which have not been properly cooled after the filling operation.

419.4. The above criteria for establishing fill limits, and the specific fill limits for the uranium hexafluoride cylinders most commonly used throughout the world are specified in ISO 7195 [26]. Fill limits for any other uranium hexafluoride cylinder should be established using these criteria and, for any cylinder requiring competent authority approval, the analysis establishing the fill limit and the value of the fill limit should be included in the safety documentation submitted to the competent authority. A safe fill limit should accommodate the internal volume of the uranium hexafluoride when in heated, liquid form and, in addition, an allowance for ullage (i.e., the gas volume) above the liquid in the container should be provided.

419.5. Uranium hexafluoride exhibits a significant expansion when undergoing the phase change from solid to liquid. The uranium hexafluoride expands from a solid at $20^{o}C$ to a liquid at $64^{o}C$ by 47% (from 0.19 cm^3/g to 0.28 cm^3/g). In addition, the liquid uranium hexafluoride will expand an additional 10%

based on the solid volume (from 0.28 cm^3/g at the triple point to 0.3 cm^3/g) when heated from 64°C to 113°C). As a result, an additional substantial increase in volume of the uranium hexafluoride between the minimum fill temperature and the higher temperatures can occur. Therefore, extreme care should be taken by the designer and the operator at the facility where uranium hexafluoride cylinders are filled to ensure that the safe fill limit for the cylinder is not exceeded. This is especially important since, if care is not taken, the quantity of material which can be added to a cylinder could greatly exceed the safe fill limit at the temperature where uranium hexafluoride is normally transferred into cylinders (e.g., at temperatures of about 71°C). For example, a 3,964 litre cylinder, with a fill limit of 12,261 kg, could accept up to 14,257 kg of uranium hexafluoride at 71°C. When heated above 71°C, the liquid uranium hexafluoride would completely fill the cylinder and could hydraulically deform and rupture the cylinder. Quantities of uranium hexafluoride above 14,257 kg would rupture the cylinder if heated above 113°C. Hydraulic rupture is a well understood phenomenon, and it should be prevented by adhering to established fill limits based on the cylinder certified minimum volume and a uranium hexafluoride density at 121°C for all cylinders or the maximum temperature relating to the design of the cylinder [27].

419.6. Prior to shipment of a uranium hexafluoride cylinder, the consignor should verify that its internal pressure is below atmospheric pressure by measurement with a pressure gauge or another suitable pressure indicating device. This is consistent with ISO 7195 which indicates that a sub-atmospheric cold pressure test should be used to demonstrate suitability of the cylinder for transport of uranium hexafluoride. According to ISO 7195, a cylinder of uranium hexafluoride should not be transported unless the internal pressure is demonstrated to be at a partial vacuum of 6.9×10^4 Pa. The operating procedure for the package should specify the maximum sub-atmospheric pressure allowed, measured in this fashion, which will be acceptable for shipment; and the results of this measurement should be included in appropriate documentation. This prior-to-shipment test should also be accomplished subject to agreed quality assurance procedures.

SECTION V

REQUIREMENTS AND CONTROLS FOR TRANSPORT

REQUIREMENTS BEFORE THE FIRST SHIPMENT

501.1. For ensuring safe transport of radioactive material, general provisions for quality assurance (para. 310) and compliance assurance (para.311) have been established in the Regulations. Specific inspection requirements to assure compliance for those packaging features which have a major bearing on the integrity of the package and on radiation and nuclear criticality safety have also been established. These requirements cover inspections both prior to the first shipment and prior to each shipment. The requirements in para. 501 relating to shielding, containment, heat transfer, and criticality safety (confinement system effectiveness and neutron poison characteristics) of specific packagings were determined to be those important design/fabrication features related to safety which need to be verified at the end of fabrication and prior to use.

501.2. In the design phase of the package, documents should be prepared to define how the requirements of para. 501 are fully complied with for each manufactured packaging. Each document required should be authorized (e.g., signed) by the persons directly responsible for each stage of manufacture. Specific values should be recorded, even when within tolerance. The completed documents should be retained on file in conformance with quality assurance requirements (see para. 310).

501.3. In the case of a containment system having a design pressure exceeding 35 kPa, as required in para. 501(a), it should be confirmed that the containment system in the as-fabricated state is sufficient. This may be accomplished, for instance, through a test. For packagings with fill/vent valves, these openings can be used to pressurize the containment system to its design pressure. If the containment system does not have such penetrations, the vessel and its closure may require separate testing using special fixtures. During these tests, seal integrity should be evaluated using the procedures established for normal use of the package.

501.4. In performing the tests and inspections on packagings following fabrication to assess the effectiveness of shielding, to satisfy para. 501(b), the shielding components may be checked by a radiation test of the completed assembly. The radiation source for this test need not be the material intended to be transported, but care should be taken such that shielding properties are properly evaluated relative to energy, energy spectrum and type of radiation. Particular attention should also be paid to the homogeneity of packaging materials and the possibility of increased localized radiation levels at joints. For methods of testing the integrity of a package's radiation shielding see Refs [28],[29] and paras 656.13 to 656.18.

501.5. Containment integrity should be assessed using appropriate leakage rate tests for compliance with para. 501(b), (see paras 656.1 to 656.12 and 656.21 to 656.24).

501.6. Inspection of a packaging for heat transfer characteristics should, in compliance with para. 501(b) in addition to a dimensional check, include special attention to ventilation apertures, surface emissivity and absorptivity and continuity of conduction paths. Proof tests, which may normally be necessary only for a prototype package, may be conducted using electrical heaters in place of a radioactive source.

501.7. Although the confinement system includes the package contents, only the packaging components of the confinement system need to be inspected and/or tested after fabrication and prior to the first shipment to comply with para. 501(b). Any inspection and/or testing of the fissile material should be performed prior to each shipment (see para. 502.2 or 501.8 as appropriate). Dimensional and material inspection of pertinent packaging components and welds should be completed to assure the confinement system packaging components are fabricated and located as designed. Testing will most often involve assurance of the presence and distribution of the neutron poisons as discussed in para. 501.8.

501.8. In cases where criticality safety is dependent on the presence of neutron absorbers as referred to in subpara. 501(c) it is preferred that the neutron absorber be a solid and an integral part of the packaging. Solutions of absorbers, or absorbers that are water soluble are not endorsed for this purpose because their continued presence cannot be assured. The confirmation procedure or tests should assure that the presence and distribution of the neutron absorber within the packaging components are consistent with that assumed in the criticality safety assessment. Merely assuring the quantity of the neutron absorbing material is not always sufficient because the distribution of the neutron absorbers within a packaging component, or within the packaging contents itself, can have a significant effect on the neutron multiplication factor for the system. Uncertainties in the confirmation technique should be considered in verifying consistency with the criticality safety assessment.

501.9. For further information see Refs[30],[31].

REQUIREMENTS BEFORE EACH SHIPMENT

502.1. In addition to the requirements imposed on certain packages prior to their first shipment (para. 501), certain other requirements are to be satisfied prior to each shipment of any package to enhance compliance and assure safety. These requirements include inspection to ensure that only proper lifting attachments are used during shipment, and verification that requirements in approval certificates are complied with and thermal and pressure stability have been demonstrated. In all cases these requirements are deemed necessary to reduce the possibility of having an unsafe package shipped in the public domain, and are aimed at prevention of human error.

502.2. Inspection and test procedures should be developed to ensure that the requirements of paras 502(a) and 502(b) are satisfied. Compliance should be documented as part of the quality assurance programme (see para. 310).

502.3. The certificate of approval (see subparas. 502(c) to (h)) is the evidence that a package design of an individual package meets the regulatory requirements and that the package may be used for transport. The provisions of para. 502 are designed to ensure that the individual package continues to comply with these requirements. Each check should be documented and authorized (e.g., signed) by the person directly responsible for that operation. Specific values should be recorded, even when within tolerances, and compared with results of previous tests, so that any indication of deterioration may become apparent. The completed documents should be retained on file in conformance with quality assurance requirements (see para. 310).

502.4. The approval certificates for packages containing fissile material indicate the authorized contents of the package (see para. 418 and para. 833). Prior to each shipment, the fissile material contents should be verified to have the characteristics provided in the listing of authorized contents. When removable neutron poisons or other removable criticality control features are specifically allowed by the certificate, inspections and/or tests, as appropriate, should be made to ascertain the presence, correct location(s) and/or concentration of those neutron poisons or control features. Solutions of absorbers, or absorbers that are water soluble are not endorsed for this purpose because their continued presence cannot be assured.

The confirmation procedure or tests should assure that the presence, correct location(s), and/or concentration of the neutron absorber or control features within the package is consistent with that assumed in the criticality safety assessment. Merely assuring the quantity of the control material is not always sufficient because the distribution within the package can have a significant effect on the reactivity of the system.

502.5. To be in compliance with para. 502(d), it is recommended that detailed procedures be developed and followed to ensure that steady state conditions have been reached by measuring the temperature and pressure over a defined period. In the performance of any test it should be ensured that the method selected does not degrade the integrity of the package and that it provides the required sensitivity. Non-conformance with the approved design requirements should be fully documented and also reported to the competent authority which approved the design.

502.6. Every Type B(U), Type B(M) and Type(C) package should be tested, after closure and before transport, to ensure compliance with the required leaktightness standard (see para. 502(e)). Some national authorities may permit an assembly verification procedure followed by a less stringent leakage test as offering equivalent confidence in meeting the design conditions. An example of an assembly verification procedure would be to:

> *First inspect and /or test comprehensively the complete containment system of an empty packaging. The radioactive contents may then be loaded into the packaging and only the closure components which were opened during loading need be inspected and/or tested as part of the assembly verification procedure.*

In the case of packages where containment is provided by radioactive material in special form, compliance may be demonstrated by possession of a certificate prepared under a quality assurance programme which demonstrates the leak tightness of the source(s) concerned. It is recommended that the competent authority of the country concerned be consulted if such a procedure is envisaged.

502.7. The leak test requirements for Type B(U), Type B(M) and Type C packages, including tests performed, frequency of testing and test sensitivity, are based on the maximum allowable leak rates and standardized leak rates calculated for the package for normal and accident conditions as described in ISO 12807 [32]. Preshipment leakage testing may not be necessary for some Type B packages depending for example on the material contained and the related allowable leak rate. Examples of such materials could include, for example, powders exceeding the specific activity limit for LSA-II material, but not qualifying as LSA-III. The physical characteristics of the material include a limited concentration of radioactivity in the material, and a physical form which reduces dispersibility of the material as discussed in 226.14 to 226.20. Other packages carrying these materials may require preshipment leak tests but these could be simple direct tests, such as gas and soap bubble qualitative tests and gas pressure drop and rise quantitative tests, as described in ISO 12807 or ANSI N.14.5 - 1977 [31].

502.8. Concerning para. 502(g), the measurement specified by 674(b) should verify that the irradiated nuclear fuel falls within the envelope of conditions demonstrated in the criticality safety assessment to satisfy the criteria of paras 671 - 682. Typically the primary conditions proposed for use in the safety assessment of irradiated nuclear fuel at a known enrichment are the burnup and decay characteristics and, as such, these are the parameters that should be verified by measurement. The measurement technique should depend on the likelihood of misloading the fuel and the amount of available subcritical margin due to irradiation. For example, as the number of fuel elements of varying irradiation stored in the reactor pond and the length of time between discharge and shipment increase, so the likelihood of misloading increases. Similarly, if an irradiation of 10 GWd/MTU is used in the criticality assessment, but fuel less than 40 GWd/MTU is not allowed to be loaded in the package, a measurement verification of irradiation using a technique with a large uncertainty may be adequate whereas if an irradiation of 35 GWd/MTU is used in the criticality assessment, a measurement technique that verifies irradiation should be much more reliable. The measurement criteria that should be met to allow the irradiated material to be loaded and/or shipped should be clearly specified in the certificate of approval. See references [33],[34],[35],[36] for information on measurement approaches in use [33] or proposed [34],[35],[36] for use.

502.9. The approval certificate should identify any requirements for closure of a package containing fissile material which are necessary as a result of the assumptions made in the criticality safety assessment relative to water inleakage for a single package in isolation (see para. 677). Inspections and/or tests should be made to ascertain any special features for prevention of water inleakage have been met.

TRANSPORT OF OTHER GOODS

504.1. The purpose of this requirement is to prevent radioactive contamination of other goods. See also paras 513.1 to 513.4 and para. 514.1.

505.1. This provision makes it possible for the consignor to include in the exclusive use consignment other goods destined to the same consignee under the conditions specified. The consignor has primary responsibility for ensuring compliance.

506.1. Dangerous goods may react with one another if allowed to come into contact. This could occur, for instance, as a result of leakage of a corrosive substance or of an accident causing an explosion. To minimize the possibility of radioactive material packages losing their containment integrity owing to the interaction of the package with other dangerous goods, they should be kept segregated from other dangerous cargo during transport or storage. The extent of segregation required is usually established by individual States or the cognizant transport organizations (International Maritime Organization (IMO), International Civil Aviation Organization (ICAO), etc.).

506.2. Information on specific storage, stowage and segregation requirements, as applicable, is contained in the transport regulatory documents of international transport organizations [6],[10],[37],[38],[39],[40] and in provisions laid down in regulatory documents of individual states. As these regulations and provisions are frequently amended, the current editions should be consulted in order to ascertain these requirements with respect to other goods.

OTHER DANGEROUS PROPERTIES OF CONTENTS

507.1. The Regulations provide an acceptable level of control of the radiation and criticality hazards associated with the transport of radioactive material. With one exception, (UF_6), the Regulations do not cover hazards which may exist due to the physical/chemical form in which radionuclides are transported. In some cases, such subsidiary hazards may exceed the radiological hazards. Compliance with the provisions of the Regulations therefore does not absolve its users from the need to consider all of the other potential dangerous properties of the contents.

507.2. This Edition of the Regulations includes, for the first time, provisions regarding packaging requirements for uranium hexafluoride (UF_6), based on both the relevant hazards, i.e., the radiological/criticality and the chemical hazards. Uranium hexafluoride is the only commodity for which such subsidiary hazards have been taken into account in the formulation of provisions in these Regulations (see para. 629).

507.3. The United Nations Recommendations on the Transport of Dangerous Goods [7] classifies all radioactive material in Class 7, though the other dangerous properties for some materials may be more onerous. The United Nations Recommendations prescribes performance tests for packagings for such goods and classifies them as follows:

Class 1 -	Explosives
Class 2 -	Gases (compressed, liquefied, dissolved under pressure or deeply refrigerated)
Class 3 -	Flammable liquids
Class 4 -	Flammable solids; substances liable to spontaneous combustion; substances which, on contact with water, emit flammable gases
Class 5 -	Oxidizing substances; organic peroxides
Class 6 -	Toxic and infectious substances
Class 7 -	Radioactive material
Class 8 -	Corrosive substances
Class 9 -	Miscellaneous dangerous substances and articles.

507.4. In addition to meeting the requirements of the Regulations for their radioactive properties, radioactive consignments must comply with the requirements specified by relevant international transport organizations and applicable provisions adopted by individual states, for each individual substance on account of its other hazardous properties. This includes, for example, requirements on labelling and information to be provided in the transport documents, and may also include additional package design requirements and approvals by appropriate authorities.

507.5. Where the packaging requirements specified by relevant international standards organizations for a subsidiary hazard are more severe than those quoted in the IAEA regulations for the radiological hazard, then the requirements for the subsidiary hazard will set the standard [7].

507.6. For radioactive material transported under pressure, or where internal pressure may develop during transport under the temperature conditions specified in the Regulations, or when the package is pressurized during filling or discharge, the package may fall under the scope of pressure vessel codes of the Member States of concern.

507.7. Performance tests for packagings of goods with dangerous properties other than radioactivity are prescribed in the United Nations Recommendations [7].

507.8. Additional labels denoting subsidiary hazards should be displayed as specified by the national and international transport regulations.

507.9. Since the regulations promulgated by the international transport organizations as well as by individual Member States are frequently amended, the current editions of them should be consulted to ascertain what additional provisions apply with respect to subsidiary hazards.

REQUIREMENTS AND CONTROLS FOR CONTAMINATION AND FOR LEAKING PACKAGES

508.1. The Regulations prescribe limits for non-fixed contamination on the surfaces of packages and conveyances. The limits for the surfaces of packages derive from a radiological model developed by

Fairbairn [41] for the 1961 Edition of the Regulations. In summary, the pathways of exposure were external beta irradiation of the skin, ingestion and the inhalation of resuspended material. Consideration of radionuclides was limited to the most hazardous radionuclides in common use, namely, ^{239}Pu and ^{226}Ra in the case of alpha emitters and ^{90}Sr in the case of beta emitters. These derived limits correspond to values that were generally accepted for laboratory and plant working areas and were thus conservative in the context of transport packages where exposure time and handling time for workers were expected to be very much less than for workers in laboratories or active plants. Since this derivation there have been changes in radiological protection principles, but the limits have not been changed because they were based on cautious models applied to the most hazardous radionuclides in common use. In the 1996 Edition of the Regulations, a radionuclide specific approach has been rejected on the grounds that it would be impracticable and unnecessary. Irrespective of the method used to determine the limit, optimization plays a role in reducing contamination levels on transport packages to levels that are as low as reasonably achievable with due regard to the dose accrued during decontamination. The existing values give rise to low doses.

508.2. In the case of packages contaminated with an alpha emitter, the pathway of exposure that usually determines a derived limit for contamination is the inhalation of material that has been resuspended from the surfaces of packages. The value of a relevant resuspension factor is uncertain but research in the field was reviewed in a report published in 1979 [42]. The wide range of reported values spans the value recommended for general use by the IAEA[43] of $5x10^{-5}$ m^{-1} which takes account of the probability that only a fraction of the activity resuspended may be in respirable form. In most cases the level of non-fixed contamination is measured indirectly by wiping a known area with a filter paper or a wad of dry cotton wool or other material of a similar nature. It is common practice to assume that the amount of activity on the wipe represents only 10% of the total non-fixed contamination present on the surface. The fraction on the wipe will include the activity most readily available for resuspension. The activity remaining in situ on the surface is less easily resuspended. An appropriate value for the resuspension factor for application to the total amount of non-fixed contamination on transport packages might be of the order of 10^{-5} m^{-1}. For an annual exposure time of 1000 h to an atmosphere polluted by activity resuspended from the surfaces of packages contaminated with ^{239}Pu at 0.4 Bq cm^{-2} and using a resuspension factor of 10^{-5} m^{-1}, the annual effective dose is about 2 mSv. In the case of contamination with ^{226}Ra the annual effective dose would be of the order of 0.1 mSv. For most beta/gamma emitters the pathway of exposure that would determine a derived limit is exposure of the basal cells of the skin. The 1990 ICRP recommendations[44] retain 7 mg cm^{-2} as the nominal depth of the basal cells, but extend the range of depth to 2-10 mg cm^{-2}. A number of studies[45],[46],[47] provide dose-rate conversion factors at a nominal depth of 7 mg cm^{-2}, or for the range 5-10 mg cm^{-2}. Skin contaminated by $^{90}Sr/^{90}Y$ at 4 Bq cm^{-2} for 8 hours per working day would give rise to an equivalent dose to the skin of about 20 mSv per year, to be compared to an annual limit of 500 mSv [2]. This assumes a transfer factor of unity between package surfaces and skin.

508.3. In practice, contamination which appears fixed may become non-fixed as a result of the effects of weather, handling, etc. In most instances where small packages are slightly contaminated on the outer surfaces, the contamination is almost entirely removable or non-fixed and the methods of measurement should reflect this. In some situations, however, such as in the case of fuel flasks which may have been immersed in contaminated cooling pond water whilst being loaded with irradiated fuel, this is not necessarily so. Contaminants such as caesium-137 may strongly adhere onto or penetrate into steel surfaces. Contamination may become ingrained in pores, and fine cracks, and in crevices, particularly in the vicinity of lid seals. Subsequent weathering, exposure to rain or even exposure to moist air conditions may cause some fixed contamination to be released or to become non-fixed. Care is necessary prior to dispatch to utilize appropriate decontamination methods to reduce the level of contamination such that the limits of non-fixed contamination would not be expected to be exceeded during the journey. It should be recognized that on some occasions the non-fixed contamination limits may be exceeded at the end of the journey. However, this situation presents no significant hazard because of the pessimistic assumptions used in calculating the derived limits for non-fixed contaminations. In such situations the consignee should inform the consignor to enable him to determine the causes and to minimize such occurrences in the future.

508.4. In all cases contamination levels on the external surface of packages should be kept as low as is reasonably achievable. The most effective way to ensure this is to prevent the surfaces from becoming contaminated. Loading, unloading and handling methods should be kept under review to achieve this. In the particular case of fuel flasks mentioned above, the pond immersion time should be minimized and effective decontamination techniques should be devised. Seal areas should be cleared by high pressure sprays, where possible, and particular care taken to minimize the presence of contaminated water between the body and lid of the flask. The use of a `skirt' to eliminate contact with contaminated water in cooling ponds can prevent contamination of surfaces of the flask. If this is not possible, the use of strippable paints, pre-wetting with clean water and initiating decontamination as soon as possible may significantly reduce contamination uptake. Particular attention should be taken to remove contamination from joints and seal areas. Surface soiling should also be avoided wherever possible. Wiping a dirty surface both removes dirt and abrades the underlying substrate, especially if the latter is relatively soft, e.g., paint or plastic. Thus soiling can contribute to non-fixed contamination either by the loose dirt becoming contaminated itself or by the action of wiping the dirty surface generating loose contamination from the underlying substrate. Paints and plastics weather on exposure to sunlight. Amongst other effects, ultraviolet light oxidizes paint or plastic surfaces, thus increasing cation exchange capacity. This renders surfaces exposed to the environment increasingly contaminable by some soluble contaminants.

508.5. It must be kept in mind that, if all packages were contaminated close to the limits, the routine handling and storage of packages in transit stores, airport terminals, rail marshalling yards, etc. could lead to buildup of contamination in working areas. It may be necessary that checks be made to ensure that such buildup does not occur. Similarly, it is advisable to occasionally check gloves or other items of clothing of personnel regularly handling packages.

508.6. The Regulations set no specific limits for the levels of fixed contamination on packages, since the external radiation resulting therefrom will combine with the penetrating radiation from the contents, and the net radiation levels for packages are controlled by other specific requirements. However, limits on fixed contamination are set for conveyances (see para. 513) to minimize the risk that it may become non-fixed as a result of abrasion, weathering, etc.

508.7. In a few cases a measurement of contamination may be made by direct reading contamination monitors. Such a measurement will include both fixed and non-fixed contamination. This will only be practicable where the level of background radiation from the installation in which the measurement is made or the radiation level from the contents does not interfere. In most cases the level of non-fixed contamination will have to be measured indirectly by wiping a known area with a smear and measuring the resultant radioactivity on the smear in an area not influenced by radiation background from other sources.

508.8. The derived limits for non-fixed contamination apply to the average level over an area of 300 cm^2 or the total package if its total surface area is less than 300 cm^2. The level of non-fixed contamination may be determined by wiping an area of 300 cm^2 by hand with a filter paper or a wad of dry cotton wool or other material of similar nature. The number of smear samples taken on a larger package should be such as to be representative of the whole surface and should be chosen to include areas known or expected to be more contaminated than the remainder of the surface. For routine surveys on a very large package such as on an irradiated fuel flask, it is common practice to select a large number of fixed general positions to assist in identifying patterns and trends. Care should be taken that exactly the same position is not wiped on each occasion since this would leave large areas never checked, and would tend to `clean' the areas checked.

508.9. The radioactivity on the smear sample may be measured either with a portable contamination monitor or in a standard counting castle. Care is necessary in converting the count rate to surface activity as a number of factors such as counting efficiency, geometrical efficiency, counter calibration and the fraction of activity removed from the surface to the smear sample will affect the final result.

508.10. To avoid underestimation, the beta energy of the calibration source used for a counter should not be greater than the beta energies of the contaminant being measured. The fraction of activity removed by the wipe test can, in practice, vary over a wide range and is dependent on the nature of the surface, the nature of the contaminant, the pressure used in wiping, the contact area of the material used for the test, the technique of rubbing (e.g., missing parts of the 300 cm^2 area or doubly wiping them) and the accuracy with which the operator estimates the area of 300 cm^2. It is common practice to assume that the fraction removed is 10% and this is usually conservative, so this results in overestimating the level of contamination. Other fractions may be used if determined experimentally.

508.11. It is recommended that users develop specific contamination measurement techniques relevant to their particular circumstances. Such techniques include the use of smears and appropriate survey instruments. The instruments and detectors selected should take into account the likely isotopes to be measured. Particular care is necessary in selecting instruments of appropriate energy dependence when low energy beta or alpha emitters are present. It should be recognized that the size of the smear and the size of the sensitive area of the detector are very important factors in determining overall efficiency.

508.12. It is further recommended that operators be adequately trained to achieve results as consistent as possible. Comparison between operators may be valuable in this respect. Attention is drawn to the difficulties which will occur if different organizations use techniques which are not fully compatible - especially in circumstances where it is not practical to maintain the levels of non-fixed contamination to near zero values.

509.1. See paras 508.1 to 508.12.

510.1. The prime purpose of inspection by a qualified person is to assess whether leakage or loss of shielding integrity has occurred or could be expected to occur, and either give assurance that the package is safe and within the limits prescribed in the Regulations or, if this is not so, to assess the extent of the damage or leakage and the radiological implications. On rare occasions it may be necessary to extend surveys and investigations back along the route, the conveyances and the handling facilities to identify and clean up any contaminated areas. Investigations may need to include the assessment of external dose and possible radioactive intake by transport workers and members of the public.

510.2. Vehicles containing damaged packages which appear to be leaking, or appear to be severely dented or breached, should be detained and secured until they have been declared safe by a qualified person.

513.1. Conveyances may become contaminated during the carriage of radioactive material from the non-fixed contamination on the packages. If the conveyance has become contaminated above this level, it should be decontaminated to a level as low as reasonably achievable and to at least the appropriate limit. This provision does not apply to the internal surfaces of a conveyance provided that the conveyance remains dedicated to the transport of radioactive material or surface contaminated objects under exclusive use. (See para. 514.1)

513.2. Limits are also set on fixed contamination to minimize the risk that it may become non-fixed as a result of abrasion, weathering, etc.

513.3. If the non-fixed contamination on conveyances exceeds the limits in para. 508 of the Regulations, the conveyance should be decontaminated and following the decontamination a measurement should be made of the fixed contamination. The radiation level resulting from the fixed contamination on the surfaces may be measured using a portable instrument of an appropriate range held near to the surface of the conveyance. Such measurements should only be made before the conveyance is loaded.

513.4. Where packages having relatively high levels of fixed contamination are handled regularly by the same transport workers it may be necessary to consider not only the penetrating radiation but also the non-penetrating radiation from that contamination. An example of such a situation is the regular transport of irradiated fuel flasks by rail, when certain transport workers may regularly come in contact with the flasks. The effective dose received by the workers from the penetrating radiation may be sufficiently low so that no special health surveillance or individual monitoring is necessary. If it is known that the fixed contamination levels may be high, then it may be prudent to derive a working limit that prevents undesirable exposure of the workers hands.

513.5. For measurement of surface dose rates, see para. 233.1 to 233.7.

514.1. While it is normally good practice to decontaminate an overpack, freight container, tank, intermediate bulk container or conveyance as quickly as possible so that it can be used for transporting other substances, there are situations, e.g., transport of uranium or thorium ores, where conveyances are essentially dedicated to the transport of radioactive materials, including unpackaged radioactive material, and are continually contaminated. In such cases where the practice of using dedicated conveyances is common, an exception to the need for quickly decontaminating these conveyances, tanks, overpacks, intermediate bulk containers or freight containers, if applicable, is provided for as long as these conveyances, tanks, overpacks, intermediate bulk containers or freight containers remain in that dedicated use. Decontamination of the internal surfaces after every use could lead to unnecessary exposure of workers. On the other hand, the external surfaces which are continually being exposed to the environment, and which are generally much easier to decontaminate, should be decontaminated to below the applicable limits after each use. It should be noted that para. 414 of the 1985 Edition of the Regulations was restricted to low specific activity materials and surface contaminated objects. This provision is now extended to apply to all radioactive material.

REQUIREMENTS AND CONTROLS FOR TRANSPORT OF EXCEPTED PACKAGES

515.1. Excepted packages are packages in which the allowed radioactive content is restricted to such low levels that the potential hazards are insignificant and therefore no testing is required with regard to containment or shielding integrity (see also paras 517.1 to 517.5).

516.1. The requirement that the radiation level at the surface of an excepted package must not exceed 5μSv/h was established in order to ensure that any radiation dose to members of the public will be insignificant and that sensitive photographic material will not be damaged.

516.2. It is generally considered that radiation exposures not exceeding 0.15 mSv do not result in unacceptable fogging of undeveloped photographic film. A package containing such film would have to remain for more than 20 hours in contact with an excepted package having the maximum radiation level on contact of 5μSv/h in order to receive the prescribed radiation dose limit of 0.1 mSv (see paras 307.1 to 307.3).

516.3. By the same argument special segregation from persons is not necessary. Any radiation dose to members of the public will be insignificant even if such a package is carried in the passenger compartment of a vehicle.

516.4. For measuring the radiation level, an appropriate instrument should be used, i.e., sensitive to and calibrated for the type of radiation to be measured. In most cases only penetrating radiation (gamma rays and neutrons) needs to be taken into account. For establishing the radiation level on the surface of a package, it is normally adequate to take the reading shown on the instrument when the instrument is held against the surface of the package. The instruments used should, where possible, be small compared with the size of the package. In view of the usually small dimension of excepted packages, instruments with a small detection chamber (Geiger-Müller tube, scintillation meter or ionization chamber) are most suited for the purpose. The instrument should be reliable, in good condition, properly maintained and calibrated, and possess characteristics acceptable in good radiation protection practice.

517.1. The limits for radioactive material contents of excepted packages are such that the radioactivity hazard associated with a total release of contents is consistent with the hazard from a Type A package releasing part of its contents (see Appendix I).

517.2. Limits other than the basic limits are allowed where the radioactive material is enclosed in or forms a component part of an instrument or other manufactured article where an added degree of protection is provided against escape of material in the event of an accident. The added degree of protection is assessed in most cases as a factor of 10, thus leading to limits for such items which are 10 times as high as the basic limits. The factor of 10 used in this and the other variations from the basic limits are pragmatically developed factors.

517.3. The added degree of protection is not available in the case of gases so that the item limits for instruments and manufactured articles containing gaseous sources remain the same as the limits for excepted packages containing gaseous material not enclosed in an instrument or article.

517.4. Packaging reduces both the probability of the contents being damaged and the likelihood of radioactive material in solid or liquid form escaping from the package. Accordingly, the excepted package limits for instruments and manufactured articles incorporating solid or liquid sources have been set at 100 times the item limits for individual instruments or articles.

517.5. With packages of instruments and articles containing gaseous sources, the packaging may still afford some protection against damage, but it will not significantly reduce the escape of any gases which may be released within it. The excepted package limits for instruments and articles incorporating gaseous sources have therefore been set at only 10 times the item limits for the individual instruments or articles.

518.1. The basic activity limit for non-special form solid material which may be transported in an excepted package is 10^{-3} A_2. This limit for an excepted package was derived on the basis of the assumption that 100% of the radioactive contents could be released in the event of an accident. The maximum radioactivity released in such an event, i.e., 10^{-3} A_2, is comparable to the fraction of the contents assumed to be released from a Type A package in the dosimetric models used for determining A_2 values (see Appendix I).

518.2. In the case of special form solid material, the probability of release of any dispersible radioactive material is very small. Thus, if radiotoxicity were the only hazard to be considered, much higher activity limits could be accepted for special form solid materials in excepted packages. However, the nature of special form does not provide any additional protection where external radiation is concerned. The limits for excepted packages containing special form material are therefore based on A_1 rather than A_2. The basic limit selected for special form solid material is 10^{-3} A_1. This limits the external dose equivalent rate from unshielded special form material to one thousandth of the rate used to determine the A_1 values.

518.3. For gaseous material, the arguments are similar to those for solid material and the basic excepted package limits for gaseous material are therefore also 10^{-3} A_2 for non-special form and 10^{-3} A_1 for special form material. It is to be noted that in the case of elemental gases the package limits are extremely pessimistic because the derivation of A_2 already embodies an assumption of 100% dispersal (see Appendix I).

518.4. Tritium gas has been listed separately because the actual A_2 value for tritium is much greater than the 40 TBq, which is an arbitrary maximum for A_2 values. The value of 2×10^{-2} A_2 is conservative in comparison with other gases even when allowing for conversion of tritium to tritiated water.

518.5. In the case of liquids, an additional safety factor of 10 has been applied because it was considered that there is a greater probability of a spill occurring in an accident. The basic excepted package limit for liquid material is therefore set at 10^{-4} A_2.

519.1. The purpose of the inactive sheath is to cover the outer surfaces of the uranium or thorium to protect them from abrasion, to absorb the alpha radiation emitted and to reduce the beta radiation level at the accessible surfaces of the article. The sheath also may be used to control the oxidation of the uranium or thorium, and the consequent buildup of non-fixed contamination on the outer surfaces of such articles.

519.2. Examples of articles manufactured from natural uranium, depleted uranium or natural thorium are aircraft counterweights made of depleted uranium and coated with an epoxy resin, and uranium encased in metal and used as a shield in packagings for X-ray and gamma ray radiography and medical treatment devices.

519.3. In the case of a depleted uranium shield incorporated in a packaging it is recommended that the uranium be sheathed with steel and the continuity of the envelope be assured by careful seam welding. As an example, the national regulations in the United States of America stipulate that the steel sheath must be at least 3.2 mm thick and the outside of the packaging is labelled showing that it contains uranium, to prevent it from inadvertently being machined or disposed of as scrap.

Additional requirements and controls for transport of empty packagings

520.1. Empty packagings which once contained radioactive material present little hazard provided they are thoroughly cleaned to reduce the non-fixed contamination levels to the excepted package levels specified in para. 508 of the Regulations, have external surface radiation levels below 5μSv/h (see para.

516) and are in good condition so that they may be securely re-sealed (see para. 520(a)); under these conditions the empty packaging may be transported as an excepted package.

520.2. The following examples describe situations where para. 520 is not applicable:

(a) An empty packaging which cannot be securely closed due to damage or other mechanical defects may be shipped by alternate means which are consistent with the provisions of the Regulations, for instance under special arrangement conditions;
(b) An empty packaging containing residual radioactive material or internal contamination in excess of the non-fixed contamination limits as specified in para. 520(c) should only be shipped as a package category which is appropriate to the amount and form of the residual radioactivity and contamination.

520.3. Determining the residual internal radioactivity within the interior of an 'empty' radioactive material packaging (see para. 520(c)) can be a difficult task. In addition to direct smears (wipes), various methods or combinations of methods which may be used include:

- gross radioactivity measurement;
- direct measurement of radionuclide; and
- material accountability, e.g., by 'difference' calculations, from a knowledge of the activity or mass of the contents and the activity or mass removed in emptying the package.

Whichever method or combination of methods is used, care should be taken to prevent excessive and unnecessary exposure of personnel during the measuring process. Special attention should be paid to high radiation levels which could exist when the containment system of an empty packaging is open.

520.4. 'Heels' of residual material tend to build up in UF_6 packagings upon emptying. These 'heels' are generally not pure UF_6 but consist of materials (impurities) which do not sublime as readily as UF_6, such as UO_2F_2, uranium daughters, fission products and transuranic elements. Steps need to be taken upon emptying to ensure the package meets the requirements of para. 520 if it is being shipped as an empty packaging; and upon refilling to ensure that radiation levels local to the `heel' are not excessively high, that the transport documents properly account for the `heel' and that the combined UF_6 contents and `heel' satisfy the appropriate material requirements. Appropriate assessment and cleaning upon either emptying or refilling may be necessary to satisfy the relevant regulatory requirements. For further information see Refs [30],[31] (see also para. 549.5).

REQUIREMENTS AND CONTROLS FOR TRANSPORT OF LSA MATERIAL AND SCO IN INDUSTRIAL PACKAGES OR UNPACKAGED

521.1. The concentrations included in the definitions of LSA material and SCO in the 1973 Edition of the Regulations were such that, if packaging were lost, allowed materials could produce radiation levels in excess of those deemed acceptable for Type A packages under accident conditions. Since industrial packages used for transporting LSA material and SCO are not required to withstand transport accidents, a provision was initiated in the 1985 Edition of the Regulations to limit package contents to the amount which would limit the external radiation level at 3 m from the unshielded material or object to 10 mSv/h. Geometrical changes of LSA material or SCO as a result of an accident are not expected to lead to an increase of this external radiation level. This limits accident consequences associated with LSA material and SCO to essentially the same level as that associated with Type A packages, where the A_1 value is based on the unshielded contents of a Type A package creating radiation levels of 100 mSv/h at a distance of 1 m.

521.2. In the case of solid radioactive waste essentially uniformly distributed in a concrete matrix placed inside a thick wall concrete packaging, the shielding of the concrete wall should not be considered as satisfying the condition of para. 521. However, the radiation level at 3 m from the unshielded concrete matrix may be assessed by direct measurement outside the thick wall of the concrete packaging and then corrected to take into account the shielding effect of the concrete wall. This method can also be used in the case of other types of packaging.

523.1. According to paras 241(a)(iii) and 523(c), SCO-I is allowed to have non-fixed contamination on inaccessible surfaces in excess of the values specified in para. 241(a)(i). Items such as pipes resulting from the decommissioning of a facility should be prepared for unpackaged transport in a way to ensure that there is no release of radioactive material into the conveyance. This can be done, for example, by using end caps or plugs at both ends of the pipes (see also para. 241.7).

524.1. The higher the potential hazards of LSA material and SCO, the greater the integrity of the package has to be. In assessing the potential hazards, the physical form of the LSA material has been taken into account.

524.2. See para. 226.1

525.1. Conveyance activity limits for LSA material and SCO have been specified, taking the potential hazards into account, including the greater hazards presented by combustible solids, contamination levels, liquids and gases in the event of an accident.

525.2. Combustible solids in Table V means all LSA-II and LSA-III materials in solid form which are capable of sustaining combustion either on their own or in a fire.

DETERMINATION OF TRANSPORT INDEX (TI)

526.1. The Transport Index (TI) is an indicator of the radiation level in the vicinity of a package, overpack, tank, freight container, conveyance, unpackaged LSA-1 or unpackaged SCO-1 and it is used in the provision of radiation protection measures during transport.

The value obtained for the TI in accordance with the following guidelines shall be rounded up to the first decimal place (e.g. 1.13 becomes 1.2) except that a value of 0.05 or less may be considered as zero.

The TI for a package is the maximum radiation level at 1m from the external surface of the package, expressed in mSv/h and multiplied by 100.

The TI for a rigid overpack or conveyance is either the maximum radiation level at 1 m from the external surface of the overpack or conveyance, expressed in mSv/h and multiplied by 100, or the sum of the TIs of all the packages contained in the overpack or conveyance.

The TI for a freight container, tank, unpackaged LSA-1 or unpackaged SCO-1 is the maximum radiation level at 1 m from the external surface of the load, expressed in mSv/h and multiplied by 100 and then further multiplied by an additional factor which depends on the largest cross-sectional area of the load. This additional multiplication factor, as specified in Table VI of the regulations, ranges from 1 up to 10. It is equal to 1 if the largest cross-sectional area of the load is 1 m^2 or less. It is 10 if the largest cross-sectional area is more than 20 m^2. The TI for a freight container may be established alternatively as the sum of the TIs of all the packages in the freight container.

The TI for a non-rigid overpack shall be determined only as the sum of TIs of all the packages in the non-rigid overpack.

The TI for loads of uranium and thorium ores and their concentrates can be determined without measuring the radiation levels. Instead, the maximum radiation level at any point 1 m from the external surface of such loads may be taken as the level specified in para 526(a). The multiplication factor of 100 and the additional multiplication factor for the largest cross-sectional area of the load are still required , when applicable as indicated above, for determining the TI of such loads.

TI limits are discussed in para. 530.1.

526.2. In the case of large dimension loads where the contents cannot be reasonably treated as a point source, radiation levels external to the loads do not decrease with distance as the inverse square law would indicate. Since the inverse square law formed the basis for the calculation of segregation distances, a mechanism was added for large dimension loads to compensate for the fact that radiation levels at distances from the load greater than 1 m would be higher than the inverse square law would indicate. The requirement of para. 526(b), which in turn imposes the multiplication factors in Table VI, provides the mechanism to make the assigned TI correspond to radiation levels at greater distances for those circumstances felt to warrant it. These circumstances are restricted to the carriage of radioactive material in tanks or freight containers and the carriage of unpackaged LSA I and SCO I. The factors approximate to those appropriate to treating the loads as broad plane sources or three dimensional cylinders [48] rather than point sources, although actual radiation profiles are more complex due to the influences of uneven self shielding, source distribution and scatter.

526.3. The TI is determined by scanning all surfaces of a package, including the top and bottom, at a distance of 1 m. The highest value measured is the value that determines the TI. Similarly, the TI for a tank, a freight container and unpackaged LSA-I and SCO-I materials is determined by measuring at 1 m from the surfaces, but a multiplication factor according to the size of the load should be applied in order to define the TI. The size of the load will normally be taken as the maximum cross-sectional area of the tank, freight container, or conveyance but where its actual maximum area is known this may be used provided that it will not change during transport.

526.4. Where there are protrusions on the exterior surface, the protrusion should be ignored in determining the 1 m distance except in the case of a finned package in which case the measurement may be made at 1 m distance from the external envelope of the package.

527.1. For rigid overpacks, freight containers and conveyances adding the TIs reflects a conservative approach as the sum of the TIs of the packages contained is expected to be higher than the TI obtained by measurement of the maximum radiation level at 1 m from the external surface of the overpack, freight container or conveyance due to shielding effects and additional distance with such measurement. In the case of non-rigid overpacks, the TI may only be determined as the sum of the TIs of all packages contained. This is necessary because the dimensions of the overpack are not fixed and radiation level measurements at different times may give rise to different results.

DETERMINATION OF CRITICALITY SAFETY INDEX (CSI)

528.1. This paragraph establishes the procedure for obtaining the criticality safety index (CSI) of a package. The value of N used to determine the CSI must be such that a package array based on this value would be subcritical under the conditions of both paras 681 and 682. It would be wrong to assume that one condition would be satisfied if the other alone has been subjected to detailed analyses. The results of any one of the specified tests could cause a change in the packaging or contents that could affect the system moderation and/or the neutron interaction between packages; thus causing a distinct change in the neutron multiplication factor. Therefore, the limiting value of N cannot be assumed to be that of normal conditions or accident conditions prior to an assessment of both conditions.

528.2. To determine N values for arrays under normal conditions of transport (see para. 681) and under accident conditions of transport (see para. 682), tentative values for N may be used. Any array of five times N packages each under the conditions specified in para. 681(b), should be tested to see if it is subcritical and any array of two times N packages each under the conditions in para. 682(b) should be tested to see if it is subcritical. If acceptable, the N can be used for determining the CSI of the package. If the assessment indicates the selected N value does not yield a subcritical array under all required conditions, then N should be reduced and the assessments of para. 681 and 682 should be repeated to assure subcriticality. Another, more thorough approach, is to determine the two N values that separately satisfy the requirements of paras 681 - 682, and then use the smaller of these two values to determine the value of the CSI. This latter approach is termed “more thorough” because it provides a limiting assessment for each of the array conditions - normal and accident.

528.3. The CSI for a package, overpack or freight container should be rounded up to the first decimal place. For example, if the value of N is 11, then 50/N is 4.5454 and that value should be rounded up to provide a CSI = 4.6. The CSI should not be rounded down. To avoid disadvantages by this rounding procedure with the consequences that only a smaller number of packages can be transported (in the given example the number would be 10) the exact value of CSI may be taken.

529.1. All packages containing fissile material, other than those excepted by para. 672, are assigned their appropriate criticality safety index (CSI) and should display the CSI value in the label of Figure 5. The consignor should be careful to confirm that the CSI for each consignment is identical to the sum of the CSI values provided on the package labels.

LIMITS ON TRANSPORT INDEX, CRITICALITY SAFETY INDEX AND RADIATION LEVELS FOR PACKAGES AND OVERPACKS

530.1. In order to comply with the general requirements for nuclear criticality control and radiation protection, limits are set for the maximum TI, the maximum CSI and the maximum external surface radiation level for packages and overpacks (see also paras 531 and 532). In the case of transport under exclusive use, these limits may be exceeded because of the additional operational controls (see also paras 221.1 to 221.6).

531.1. See para. 530.1.

532.1. See para. 530.1.

532.2. Even though a package is permitted to have an external radiation level up to 10 mSv/h the requirements for a maximum dose limit of 2 mSv/h on the surface of the conveyance or of 0.1 mSv/h at any point 2 m from the surface of the conveyance (see para. 566) may be more limiting in certain instances. See also para. 233.3 regarding the build up of daughter nuclides in transport.

CATEGORIES

533.1. All packages, overpacks, freight containers and tanks other than those consisting entirely of excepted packages must be assigned a category. This is a necessary prerequisite to labelling and placarding.

533.2. Packages, overpacks, freight containers and tanks other than those consisting entirely of excepted packages must be assigned to one of the categories I-WHITE, II-YELLOW or III-YELLOW to assist in handling and stowage. The applicable category is determined by the TI and the radiation level at any point on the external surface of the package or overpack. In certain cases the upper limit for the TI, and for the surface radiation level, may be in excess of what would normally be allowed for packages or overpacks in the highest category, i.e., III-YELLOW. In such cases the Regulations require that the consignment be transported under exclusive use conditions.

533.3. The radiation level limits inherent in the definition of the categories have been derived on the basis of assumed package/cargo handling procedures, exposure times for transport workers and exposure times for photographic film. Historically these were derived as follows [16]:

(a) 0.005 mSv/h at surface

This surface limit was derived, not from consideration of radiation effects on man, but from the more limiting effect on undeveloped photographic film. Evaluation of the effect of radiation on sensitive x-ray film in 1947 showed that threshold fogging would occur at an exposure of 0.15 mSv and a limit was set in the 1961 Edition of the Regulations of 0.1 mSv linked to a nominal maximum exposure time of 24 hours. In later editions of the Regulations (1964, 1967, 1973 and 1973 (As Amended)), the 24 hour period was rounded to 20 hours and the limiting dose rate of 0.005 mSv/h was taken as a rounded-down value to provide protection to undeveloped film for such periods of transport. This dose rate was applied as a surface limit for category I-WHITE packages which would ensure there being little likelihood of radiation damage to film or unacceptable doses to transport personnel, without need for segregation requirements.

(b) 0.1 mSv/h at 1 m

For the purposes of limiting the radiation dose to film and man the dose of 0.1 mSv discussed in (a) above was combined with the exposure rate at 1 m from the package and an exposure time of one hour to give the 10 times TI limitation of the 1964, 1967 and 1973 Editions of the Regulations (10 'radiation units' in the 1961 Edition). This was based upon an assumed transit time of 24 hours and the conventional separation distance of 4.5 m (15 feet) between parcels containing radium in use by the US Railway Express Company in 1947. The above limitation would yield a dose of 0.115 mSv at 4.5 m (15 feet) in 24 hours.

(c) 2.0 mSv/h at surface

A separate limit of 2.0 mSv/h at the surface was applied in addition to the limit explained in (b) above on the basis that a transport worker carrying such packages for 30 minutes a day, held close to the body, would not exceed the then permissible dose of 1 mSv per 8 hour working day.

While such doses would no longer be acceptable, the adequacy of the current radiation level limits, in terms of radiological safety, has been confirmed by a number of surveys where radiation exposure of transport workers has been determined [49],[50],[51],[52] and by an assessment performed by the IAEA in 1985 [53]. However, it is recognized that the permitted radiation levels around packages and conveyances do not alone ensure acceptably low doses and the Regulations also require the establishment of radiation protection programmes (para. 301) and the periodic assessment of radiation doses to persons due to the transport of radioactive material (para. 304).

MARKING, LABELLING AND PLACARDING

Marking

534.1. In order to retain the possibility to identify the consignee or consignor of a package for which normal control is lost (e.g., lost in transit or misplaced), an identification marking is required on the package. This marking may consist of the name or address of either the consignor or consignee, or may be a number identifying a way-bill or transport document which contains this information.

534.2. See also paras. 536.2 to 536.6 for general advice on compliance with the requirement for the marking to be legible and durable.

535.1. The United Nations numbers, each of which is associated with a unique proper shipping name, have the function of identifying dangerous goods, either as specifically named substance or in generic groups of consignments. The UN numbers for radioactive material were agreed through joint international co-operation between United Nations Group of Experts and the IAEA. The system of identification by means of numbers is preferable to other forms of identification using symbols or language due to their relative simplicity in terms of international recognition. This identification can be used for many purposes. UN numbers which are harmonized with other dangerous goods permit rapid and appropriate identification of radioactive goods with secondary dangerous goods characteristics. Another example is the use of the United Nations numbers as a unique identification for emergency response operations. Each UN number can be associated with a unique emergency response advice table which permits first responders to refer to general advice in the unavoidable absence of a specialist. During the first stages of an emergency, this prepared information can be more easily accessible to a wide group of non-specialist emergency responders. (See also paras 547.1 and 549.1 to 549.5)

535.2. UN numbers for radioactive material are now used to relate to IAEA international regulatory requirements in the Schedules to the Regulations. This has proven to be an advantage in terms of identifying the applicable requirements. UN numbers can also be used for compliance situations for performance checks and controls, data collection and other statistical purposes should the competent authority find merit in this application.

535.3. See also paras. 536.2 to 536.6 for general advice on compliance with the requirement for the marking to be legible and durable.

536.1. Packages exceeding 50 kg gross mass are likely to be handled by mechanical rather than manual means and require marking of the gross mass to indicate the possible need for mechanical handling, and observance of floor loading and vehicle loading limits. In practice however, even packages having a gross mass of 50 kg should not regularly be handled manually. Before packages are handled manually on a regular basis, a procedure should be available to ensure that the radiological consequences are as low as reasonably achievable (ALARA) (see para. 301). Mechanical means should be used wherever practicable. To be useful in this respect, the marking is required to be legible and durable.

536.2. Markings on packages should be boldly printed, of sufficient size and sensibly located to be legible, bearing in mind the likely handling means to be employed. A character height of 12.5 mm should be considered a suitable minimum for light weight packages (i.e., up to a few hundred kilograms) where close contact by mechanical means, e.g., forklift trucks, are likely to be used. Heavier packages will require more ‘remote’ handling methods and the character size should be increased accordingly to allow operators to read them at a distance. A size of 65 mm is considered to be sufficient for the largest packages of tens of tonnes to the hundred tonne range. To ensure legibility, a contrasting background should be applied prior to marking if the external finish of the package does not already provide a sufficient contrast. Black characters on a white background are suitable. Where packages have irregular outer surfaces (e.g., fins or corrugations), or surfaces unsuitable for direct application of the markings, it may be necessary to provide a flat board or plate on which to place the markings to enhance legibility.

536.3. Markings should be durable in the sense of being at least resistant to the rigours of normal transport, including the effects of open weather exposure and abrasion, without substantial reduction in effectiveness. Attention is drawn to the need to consult national and modal transport regulations which may contain stricter requirements. For example, the International Maritime Dangerous Goods (IMDG) Code [6] requires all permanent markings (and also labels) to remain identifiable on packages surviving at least three months immersion in the sea. When a board or plate is used to bear a marking, it should be fitted securely to the package in a manner which is consistent with the integrity standard of the package itself.

536.4. The means of marking will depend on the nature of the external surface of the packaging itself, ranging (in order of durability) from a printed label (for the name of the consignee or consignor, UN number and proper shipping name or the gross mass), stencilling or soft stamping with indelible inks or paints (suitable for fibreboard or wooden packagings), through branding (for wooden packagings), painting with enamel or resin based paints (suitable for many surfaces, particularly metals), to hard stamping, embossing or `cast-in' markings of metallic outer packagings.

536.5. Appropriate national and modal transport regulations should always be consulted to supplement the general advice in paras 536.2 to 536.4 as variations in detailed requirements may be considerable.

536.6. The scheduled inspection and maintenance programme required for packagings should include provisions to inspect all permanent markings and to repair any damage or defects. Experience from such inspections will indicate whether durability has been achieved in practice.

537.1. The 1996 Edition of the Regulations introduces the requirement to identify Industrial packages with a mark. The design of the mark is consistent with other similar marks in that it includes the word "Type" together with the appropriate Industrial Package description (e.g., Type IP-2). The design of the mark also avoids potential confusion where, in other transport regulations, the abbreviation IP may be used for a different purpose. For example, the ICAO Technical Instructions use "IP" to mean Inner Packaging; e.g., "IP.3" to denote one out of ten particular kinds of inner packagings.

537.2. Although no competent authority approval is required for Industrial Type packages whose contents are not fissile material, the designer and/or consignor should be in a position to demonstrate compliance to any cognizant competent authority. This marking assists in the inspection and enforcement activities of the competent authorities. The marking would also provide, to the knowledgeable observer, valuable information in the event of an accident.

537.3. See also paras 536.2 to 536.6 for general advice on compliance with the requirement for the marking to be legible and durable.

538.1. All Type B(U), Type B(M), Type C and fissile material package designs require competent authority approval. Markings on such packages aim at providing a link between the individual package and the corresponding national competent authority design approval (identification mark), as well as information on the kind of competent authority design approval. Furthermore, the marking of the package provides, to the knowledgeable observer, valuable information in the event of an accident. In the case of package designs for uranium hexafluoride, the requirement for packages to bear a specific competent authority identification mark as provided in para. 828(c) depends upon the commencement of liability to receive competent authority approval, the due dates for which are given in para. 805.

538.2. The marking of a serial number is required because operational quality assurance and maintenance activities are oriented towards each packaging and the corresponding need to perform and verify these activities on an individual packaging basis. The serial number is also necessary for the competent authority's compliance assurance activities and for application of paras 815, 816 and 817.

538.3. General advice on legibility, durability of marking methods and inspection/maintenance of markings is given in paras 536.2 to 536.4. However, where possible the competent authority identification mark, serial number and Type B(U), Type B(M), or Type C mark should not be rendered illegible, obliterated or removed even under accident conditions. It may be convenient to apply such markings adjacent to the trefoil symbol on the external surface of the package. For example an embossed metal plate may be used to combine these markings.

538.4. An approved package design may be such that different internal components be used with a single outermost component, or the internal components of the packaging may be interchangeable between more than one outermost component. In these cases, each outermost component of the packaging with a unique serial number will identify the packaging as an assembly of components which satisfies the requirements of para. 538 (b), provided that the assembly of components is in accordance with the design approved by the competent authorities. In such cases, the quality assurance programme established by the consignor should ensure the correct identification and use of these components.

539.1. The marking of a Type B(U), Type B(M) or Type C package with a trefoil symbol resistant to the effects of fire and water is intended to ensure that such type of package can be positively identified as carrying radioactive material after a severe accident.

540.1. LSA-I and SCO-1 material may be transported unpackaged under the specifications given in para. 523. One of the conditions specified sets out to ensure that there will be no loss of contents during normal conditions of transport. Depending on characteristics of the material, wrapping or similar measures may be suitable to satisfy this requirement. Wrapping may also be advantageous from a practical point of view, for example to be able to affix a label to carry information of interest to the consignee or consignor. In situations where it is desirable to clearly identify the consignment as carrying radioactive material, the Regulations explicitly allow such an identifier to be marked on the wrapping or receptacle. It is important to note that the Regulations do not require such marking; the option is, however, made available for application where it is considered useful.

Labelling

541.1. Packages, overpacks, tanks and freight containers can be characterized as handling or cargo units. Transport workers need to be made aware of the contents when such units carry radioactive materials and need to know that potential radiological and criticality hazards exist. The labels provide that information by the trefoil symbol, the colour and the category (I-WHITE, II-YELLOW or III-YELLOW) and the fissile label. Through the labels it is possible to identify (a) the radiological or criticality hazards associated with the radioactive content of the cargo unit and (b) the storage and stowage provisions which may be applicable to such units.

541.2. The radioactive material labels used form part of a set of labels used internationally to identify the various classes of dangerous goods. This set of labels has been established with the aim of making dangerous goods easily recognizable from a distance by means of symbols. The specific symbol chosen to identify cargo units carrying radioactive material is the trefoil (see para. 539 and Fig. 1 of the Regulations).

541.3. The content of a cargo unit may, in addition to its radioactive properties, also be dangerous in other respects, e.g., corrosive or flammable. In these cases the regulations pertaining to this additional hazard must be adhered to. This means that, in addition to the radioactive material label, other relevant labels need to be displayed on the cargo unit.

542.1. For tanks or freight containers, because of the chance that the container could be obscured by other freight containers and tanks, the labels need to be displayed on all four sides in order to ensure that a label is visible without people having to search for it and to minimize the chance of its being obscured by other units or cargo.

Labelling for radioactive contents

543.1. In addition to identifying the radioactive properties of the contents the labels also carry more specific information regarding the contents, i.e., the name of the nuclide, or the most restrictive nuclide in the case of a mixture of radionuclides, and the corresponding activity. This information is important in the event of an incident or accident where content information may be needed to evaluate the hazard. The more specific information regarding the contents is not required for LSA-I material, because of the low radiation hazard associated with such material.

543.2. Yellow labels also show the TI of the cargo unit (i.e., package, overpack, tank and freight container). The TI information is essential in terms of storage and stowage in that it is used to control the accumulation and assure proper separation of cargo units. The Regulations prescribe limits on the total sum of TIs in such groups of cargo units. (See Table IX of the Regulations, for transport not under exclusive use.)

543.3. In the identification of the most restrictive radionuclides for the purpose of identifying a mixture of radionuclides as the contents on a label, consideration should be given not only to the lowest A_1 or A_2 values, but also to the relative quantities of radionuclides involved. For example, a way to identify the most restrictive radionuclide is by determining for the various radionuclides the value of

$$\frac{f_i}{A_i}$$

where f_i = activity of radionuclide i , and
$A_i = A_1$ or A_2 for radionuclide i as applicable

The highest value represents the most restrictive radionuclide.

Labelling for criticality safety

544.1. The criticality safety index (CSI) is a number used to identify the control needed for criticality safety purposes. The control is provided by limiting the sum of the CSI to 50 for shipments not under exclusive use and to 100 for shipments under exclusive use.

544.2. The labels carrying the criticality safety index (CSI) should appear on packages containing fissile material, as required by para. 541. The CSI label is additional to the category labels (Categories I-WHITE, II-YELLOW AND III-YELLOW), because its purpose is to provide information on the CSI, whereas the Category label provides information on the transport index (TI) and the contents. The CSI label, in its own right, also identifies the package as a Fissile Package.

544.3. Like the TI, the CSI provides essential information relevant to storage and stowage arrangements in that it is used to control the accumulation and assure proper separation of cargo units with fissile material contents. The Regulations prescribe limits on the total sum of CSIs in such groups of cargo units. (See Table X of the Regulations for both transport under and not under exclusive use).

545.1. See paras 544.1 to 544.3.

Placarding

546.1. Placards, which are used on large freight containers and tanks (and also road and rail vehicles -see para. 570), are designed in a similar way to the package labels (although they do not bear the detailed information of TI, contents and activity) in order to clearly identify the hazards of the dangerous goods. Displaying the placards on all four sides of the freight containers and tanks ensures ready recognition from all directions. The size of the placard is intended to make it easy to read, even at a distance. To prevent the need for an excessive number of placards and labels, an enlarged label only may be used on large freight containers and tanks where the enlarged label also serves the function of a placard.

547.1. The display of the United Nations Number can provide information on the type of the radioactive material transported including whether or not it is fissile, and information about the package type. This information is important in the case of incidents or accidents resulting in leakage of the radioactive material in that it assists those responsible for emergency response to determine proper response actions (see para.535.1).

CONSIGNOR'S RESPONSIBILITIES

Particulars of consignment

549.1. The list of information provided by the consignor in complying with para. 549, is intended to inform the carrier and the consignee as well as other parties concerned, on the exact nature of a consignment, so that all appropriate actions may be taken. In providing this information, the consignor is also, incidentally, reminded of the regulatory requirements applicable to the consignment throughout its preparation for transport and on despatch (see also para. 535.1).

549.2. A list of the proper shipping names and the corresponding United Nations Numbers has been included in Table VIII of the Regulations.

549.3. The attention of the consignor is drawn to the particular requirement of para. 549(k) regarding consignments of packages in an overpack or freight container. Each package or collection of packages is required to have documents for the appropriate consignee. This is important in regard to the `Consignor's declaration'. Nobody other than the consignor can make this declaration and so he is required to assure that appropriate documents are prepared for all parts of a mixed consignment so that they can continue their journey after being removed from an overpack or freight container.

549.4. Care should be exercised in the selection of the proper shipping name from the Table VIII of the Regulations. Portions of an entry not highlighted by capital letters are not considered part of the proper shipping name. When the proper shipping name contains the conjunction "or", then only one of the possible alternatives should be used. The following examples illustrate the selection of proper shipping names of the entry for UN Nos. 2909, 2915 and 3332:

UN No. 2909 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE, ARTICLES MANUFACTURED FROM NATURAL URANIUM or DEPLETED URANIUM or NATURAL THORIUM

The proper shipping name is the applicable description from the following:

UN No. 2909 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE, ARTICLES MANUFACTURED FROM NATURAL URANIUM
UN No. 2909 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE, ARTICLES MANUFACTURED FROM DEPLETED URANIUM
UN No. 2909 RADIOACTIVE MATERIAL, EXCEPTED PACKAGE, ARTICLES MANUFACTURED FROM NATURAL THORIUM

UN No. 2915 RADIOACTIVE MATERIAL, TYPE A PACKAGE, non-special form, non-fissile or fissile-excepted
UN No. 3332 RADIOACTIVE MATERIAL, TYPE A PACKAGE, SPECIAL FORM, non-fissile or fissile-excepted

The proper shipping name is the applicable description from the following:

UN No. 2915 RADIOACTIVE MATERIAL, TYPE A PACKAGE
UN No. 3332 RADIOACTIVE MATERIAL, TYPE A PACKAGE, SPECIAL FORM

As can be seen from the example UN No. 3332, the added characteristic (here Special Form) is explicitly spelled out.

549.5. Another example related to the interpretation and use of UN Number concept relates to empty packagings which have contained radioactive material; i.e., UN No. 2908. If there are residues or `heels' in the packaging, e.g., in $UF_6$ packages, the packaging should not be called `empty packaging' but should be shipped as a package (i.e., not as a packaging). The quantity remaining would determine the package category (see also para. 520.4).

549.6. The maximum activity of the contents during transport is required to be included in the transport documents (para. 549(f)). In some cases the activity may increase as a result of the buildup of daughter nuclides during transport. In such cases a proper correction should be applied in order to determine the maximum activity.

549.7. Advice on the identification of the most restrictive nuclides is given in para. 543.3. Appropriate general descriptions may include, when relevant, irradiated (or spent) nuclear fuel or specified types of radioactive waste.

549.8. It is necessary for SCO-I and SCO-II to indicate the total activity as a multiple of A_2. The activity should be calculated from the surface contamination and the area. In the case that the nuclide cannot be identified, the lowest A_2 value among the possible a nuclides and the ß-? nuclides should be used for the calculation of the total activity.

Removal or covering of labels

554.1. The purpose of labels is to provide information on the current package contents. Any previously displayed label would give the wrong information.

TRANSPORT AND STORAGE IN TRANSIT

Possession of certificates and instructions

561.1. As well as having a copy of the package approval certificate in their possession, the consignor should also ensure that he is in compliance with the requirements of the Regulations. In some countries it may be necessary for the consignor to register as a user of that certificate with the appropriate competent authority.

Segregation during transport and storage in transit

562.1. Specific attention has been drawn to the need for segregation in transport and storage in transit to ensure that radiation exposures to persons and undeveloped photographic film remain in accordance with the principles of paras 306 and 307. Section V deals with controls during transport, and in this context it is necessary to take specific steps to ensure that the principles are translated into requirements with which carriers can easily comply. The Regulations do not specifically do this since the conditions of carriage are very dependent on the mode of transport; the international transport organizations are in a better position to prescribe specific requirements and to reach the appropriate audience.

562.2. In order to implement the principles for radiation protection contained in paras 301 to 307, simple procedures have been developed which will suitably limit radiation exposures to both persons and undeveloped film.

562.3. An effective way of limiting exposures to persons during transport is to require appropriate segregation distances between the radioactive material and the areas where people may be present. The Regulations provide the basis for the determination of segregation requirements but the actual determination and specification of these requirements is done at the modal level. Segregation distance requirements are prescribed by national regulatory bodies and international transport organizations such as the International Civil Aviation Organization (ICAO) [37], the International Maritime Organization (IMO) [6], etc. They have been derived on the basis of radiological models and confirmed by experience: actual doses arising from the use of these distances in the air and sea mode have been very much lower than the limiting values of dose originally used in the models which derived them. In addition, in the requirements of ICAO [37] and IATA [39] care should be taken with state, airline and operator variations which may be more restrictive than the provisions contained in the IAEA Regulations.

562.4. There are many transport mode specific considerations and conditions which should be factored into the models used to calculate segregation distances. These include consideration of how the relationship between accumulated transport indices in a location and radiation levels in occupied areas are affected by shielding and distance and how exposure times for workers and members of the public depend upon the frequency and duration of their travel in conjunction with radioactive material. These may be established by programmes of work using questionnaires, surveys and measurements. In some circumstances exposure for a short time close to packages, for example during inspection or maintenance work on sea voyages, can be more important than longer exposure times in more regularly occupied areas. An example of the use of a model for determining minimum segregation and spacing distances for passenger and cargo aircraft is given in Appendix III.

TABLE II. SAMPLE SEGREGATION BETWEEN CLASSES
(Taken from the IMDG-Code, page 0129, Amendment 27.94)

562.5. Inevitably such calculations will be based on assumptions which may differ from real parameters in particular circumstances. Models should be robust and conservative. However, those that use all "worst case" parameters may result in recommendations leading to unnecessary practical difficulties or financial penalties. That the application of the resulting segregation distances leads to acceptably low doses, is more important than the basis on which the distances were calculated. However, transport patterns are subject to change and doses should be kept under review.

562.6. The virtues of simplicity should not be ignored. Clear and simple requirements are more easily, and more likely to be followed, than complex more rigorous ones. The simplified segregation table in the IMDG code [6] giving practical segregation distances for different vessel types and the translation of the segregation distances of ICAO's Technical Instructions [37] by operators into TI limits per hold are good examples of this.

562.7. When calculating segregation distances for storage transit areas, the TI of the packages and the maximum time of occupancy should be considered. If there is any doubt regarding the effectiveness of the distance then a check may be made using appropriate instruments for the measurement of radiation levels.

562.8. If different classes of dangerous goods are being transported together, there is a possibility that the contents of leaking packages may affect adjacent cargo, e.g., a leak of corrosive material could reduce the effectiveness of the containment system for a package of radioactive material. Thus, in some cases it has been found necessary to restrict the classes of dangerous goods that may be transported near other classes. In some cases it may simply be stated which classes of dangerous goods must be segregated from others. In order to provide a complete and easy procedure for understanding the requirements it has been found that presentation of this information in a concise tabular form is useful. As an example of a segregation table, the one included in Section 15 of the General Introduction to the IMDG-Code [6] is given in Table II.

562.9. Since mail bags often contain undeveloped film and will not be identified as such, it is prudent to protect mail bags in the same way as identified undeveloped film.

Stowage during transport and storage in transit

564.1. The retention of packages within or on conveyances is required for several reasons. By virtue of the movement of the conveyance during transport, small packages may be thrown or may tumble within or on their conveyances if not retained, resulting in their damage. Packages may also be dropped from the conveyance, resulting in their loss or damage. Heavy packages may shift position within or on a conveyance if not properly secured, which could make the conveyance unstable and could thereby cause an accident. Packages should also be restrained to avoid their movement in order to ensure that the radiation dose rate on the outside of the conveyance, to the driver or to the crew is not increased.

564.2. Within the context of the Regulations, “stowage” means the locating within or on a conveyance of a package containing radioactive material relative to other cargo (both radioactive and non-radioactive), and “retention” means the use of dunnage, braces, blocks, or tie-downs, as appropriate, to prevent movement within or on a conveyance during routine transport. When a freight container is used either to facilitate the transport of packaged radioactive material or to act as an overpack, consideration should be made for the packages to be restrained within the freight container. Methods of retention, e.g., lashings, throw-over nets or compartmentation, should be used to prevent damage to the packages when the freight container is being handled or transported.

564.3. For additional guidance on the methods of retention, see Appendix V.

565.1. Some Type B(U), Type B(M) and Type (C) packages of radioactive materials may give off heat. This is a result of radiation energy being absorbed in the components of the package as heat which is transferred to the surface of the package and thence to the ambient air. In such cases, heat dissipation capability is designed into the package and represents a safe and normal condition. For example, cobalt 60 produces approximately 15W per 40 TBq. Since most of this is absorbed in the shielding of the package, the total heat load can be in the order of 1000's of watts. The problem can be compounded if there are several similar packages in the shipment. As well as paying attention to the materials next to the packages, care should be taken to ensure that the air circulation in any compartment containing the packages is not overly restricted to cause a significant increase in the ambient temperature immediately in the area of the packages. Carriers must be careful not to reduce the heat dissipation capability of the package(s) by covering the package(s) or overstowing or close-packing with other cargo which may act as thermal insulation. When packages of radioactive materials give off significant heat, the consignor is required to provide the carrier with instructions on the proper stowage of the package.

565.2. Studies have shown that if the rate of generation of heat within a package is small (corresponding to a surface heat flux of less than 15 W/m^2), it can be dissipated by conduction alone and the temperature will not exceed 50^{o}C even if the package is completely surrounded by bulk loose cargo. The air gaps that always exist between packages allow sufficient dissipation to occur by air convection.

566.1. There are two primary reasons for limiting the accumulation of packages in groups, or in conveyances and freight containers. When packages are placed in close proximity to each other, control must be exercised to:

(a) Prevent the creation of higher than acceptable radiation levels as a result of the additive effects of radiation from the individual packages. For consignments not carried under exclusive use this is done by placing a limit on the total number of transport indices. The theoretical maximum dose rate at 2 m from the surface of a vehicle carrying 50 transport indices was historically calculated as 0.125 mSv/h, and considered to be equivalent to 0.1 mSv/h since the maximum was unlikely to be reached. Experience has confirmed the acceptability of these values.

(b) Prevent nuclear criticality by limiting neutron interaction between packages containing fissile material. Restriction of the sum of the CSIs to 50 in any one group of packages (100 under exclusive use) and the 6 m spacing between groups of packages provide this assurance.

566.2. It should be noted that for the transport of a freight container there may be more than one entry in Table IX or Table X of the Regulations respectively, which may be applicable. As an example, for a large freight container to be carried on a seagoing vessel there is no limit on the number of TIs or CSIs as regards the total vessel whereas there is a limitation of 200 TIs or CSIs in any one hold, compartment or defined area. It is also important to note that several requirements presented in the footnotes apply to certain shipments. These footnotes are requirements and are not just for information.

567.1. Any consignment having a criticality index (CSI) greater than 50 is also required to be transported under exclusive use (see para. 530.1). The loading arrangement assumed in the criticality assessment of paras 681-682 consists of an arrangement of identical packages. A study by Mennerdahl [54] provides a discussion of theoretical packaging arrangements that mix the package designs within the array and indicate the possibility for an increase in the neutron multiplication factor in comparison to an arrangement of identical packages. Although such arrangements are unlikely in practice, care should be taken in establishing the loading arrangement for shipments where the CSI exceeds 50. Boudin [55] proposes some guidance for assuring that packages of mixed design are properly arranged in an array to maintain a safe configuration. Where the CSI for a shipment exceeds 50 there is also a requirement to obtain a shipment approval (see para. 820).

Segregation of packages containing fissile material during transport and storage in transit

568.1. The requirement to maintain a spacing of 6 m is necessary for nuclear criticality control. Where two storage areas are divided by a wall, floor, or similar boundary, storage of the packages on opposite sides of the separating physical boundary has still to meet the requirement for 6m segregation.

569.1. See 568.1.

Additional requirements relating to transport by rail and by road

570.1. See paras 546.1 and 547.1.

570.2. Vehicles qualifying for the reduced size of placard would normally be less than a permissible gross mass of 3,500 kg.

571.1. See para. 547.1.

572.1. See paras 221.1 to 221.6 on exclusive use.

572.2. In most cases the radiation level at any point on the external surface of a package is limited to 2 mSv/h. For road and rail transport, when transported under exclusive use, packages and overpacks are allowed to exceed 2 mSv/h if access to the enclosed areas in the vehicle is restricted. Restricting of access to these areas may be achieved by using an enclosed vehicle that can be locked, or by bolting and locking a cage over the package. In some cases the open top of a vehicle with side walls may be covered with a tarpaulin but this type of enclosure would generally not be considered adequate for preventing access.

572.3. During transit there should be no unloading or entering into the enclosed area of a vehicle. If the vehicle is being held in the carrier's compound for any period it should be parked in an area where access is controlled and where people are not likely to remain in close proximity for an extended period. If it is required to do maintenance work on the vehicle for an extended period, then arrangements should be made with the consignor or the consignee to ensure adequate radiation protection, e.g., by providing extra shielding and radiation monitoring.

572.4. It is essential to secure a package or overpack to prevent movement during transport which could cause the radiation level to exceed relevant limits or to increase the dose to the vehicle driver. For road transport a package or overpack should be secured for acceleration, braking, and turning as expected during normal conditions of transport. For rail transport, packages should also be secured to prevent movement during `humping' of the rail car. See paras 564.1 to 564.5.

572.5. In establishing the dose rate for a conveyance, account may be taken of additional shielding within the conveyance. However, the integrity of the shielding should be maintained during routine transport; otherwise the conveyance radiation limit may not be met.

572.6. While it is a condition of para. 572(a)(iii) of the Regulations for exclusive use shipments that there must be no loading or unloading during the shipment, this does not preclude a carrier, who is consolidating consignments from more than one source to assume the role and responsibility of the consignor for a combined consignment and being so designated for the purpose of the subsequent exclusive use shipment.

573.1. The restrictions upon who may be permitted in vehicles carrying radioactive packages which may have significant radiation levels are to prevent unnecessary or uncontrolled exposures of persons.

573.2. The term "assistants" should be interpreted as meaning any worker, being subject to the requirements of para. 305, whose business in the vehicle concerns either the vehicle itself or the radioactive consignment. It could not, for example include any members of the public or passengers in the sense of those whose sole purpose in the vehicle is to travel. It could, however, include an inspector or health physics monitor in the course of his or her duties.

573.3. Vehicles should be loaded in such a way that the radiation level in occupied positions is minimized. This may be achieved by placing packages with higher radiation levels furthest away from the occupied area and placing heavy packages with low radiation levels nearer to the occupied position. During loading and unloading direct handling times should be minimised and the use of handling devices such as nets or pallets should be considered in order to increase the distance of packages from the body. Personnel should not linger unnecessarily in areas where significant radiation levels exist.

573.4. There was a provision concerning the radiation level at any normally occupied position in the case of road vehicles in the 1985 edition of Safety Series No. 6. This provision was deleted in the 1996 edition of the IAEA regulations. It has effectively been superseded by the introduction of the concept of Radiation Protection Programmes (see paras 301 and 305).

Additional requirements relating to transport by vessels

574.1. Each mode of transport has its own unique features. In the case of transport by sea the possibility of journey times of weeks or months and the need for continued routine inspection throughout the journey might lead to significant exposures during the carriage of the radioactive material. Simply having the exclusive use of a hold, compartment or defined deck area, particularly the latter, was not felt to provide sufficient radiological control for high radiation level packages. Two further restrictions were therefore introduced for packages having a surface radiation level greater than 2 mSv/h: either they must be in (or on) a vehicle or they must be transported under special arrangement. Access, and radiation levels are therefore controlled by the provisions of para. 572 for vehicles or by controls relevant to particular circumstances prescribed by the competent authority under the terms of the special arrangement.

574.2. Transport by sea of any package having a surface radiation level exceeding 2 mSv/h is only allowed under special arrangement conditions, except when transported in or on a vehicle under exclusive use and when subject to the conditions of para. 572. However, if the latter situation occurs, it may be desirable for purposes of radiation protection that a specific area be allocated for that vehicle by the master of the ship or the competent authority concerned. This would be appropriate in particular for the transport of such vehicles aboard roll-on/roll-off ships such as ferries. Further guidance will be found in the IMDG Code [6].

575.1. The simple controls on the accumulation of packages as a means of limiting radiation exposure (para. 566) may not be appropriate for ships which are dedicated to the transport of radioactive material. Since the vessel itself may be transporting consignments from more than one consignor, it could not be considered as being under exclusive use and the requirements of Tables IX and X might therefore be unnecessarily restrictive.

575.2. Special use vessels which are employed for the transport by sea of radioactive material have been adapted and/or dedicated specifically for that purpose. The required radiation protection programme should be based upon preplanned stowage arrangements specific to the vessel in question and to the number and the nature of the packages to be carried. The radiation protection programme should take into account the nature and intensity of the external radiation likely to be emitted by packages; occupancy factors based on the planned maximum duration of voyages should also be taken into account. This information should be used to define stowage locations in relation to regularly occupied working spaces and living accommodation, in order to ensure adequate radiological protection of persons. The competent authority, normally the competent authority of the flag state of the vessel, may specify the maximum number of packages permitted, their identity and contents, the precise stowage arrangements to be observed and the maximum radiation levels permitted at key locations. The radiation protection programme would normally require that appropriate monitoring be carried out during and after completion of stowage as necessary to ensure that specified doses or dose rates are not exceeded. Details of the results of such surveys, including any checks for contamination of packages and of cargo spaces may have to be provided to the competent authority.

575.3. For packages containing fissile material, the programme will also need to take appropriate account of the need for nuclear criticality control.

575.4. Although not directly part of a radiation protection programme, limitations on stowage associated with the heat output from each package may need to be considered. The means for heat removal, both naturally and mechanically, will need to be assessed for this purpose and heat outputs for individual packages need to be specified.

575.5. Records of measurements taken during each voyage may be required to be supplied to the competent authority on request. This is one method of assuring that the radiation protection programme and any other controls have functioned adequately.

575.6. "Persons qualified in the carriage of radioactive material" should be taken to mean persons who possess appropriate special knowledge of the handling of radioactive material.

575.7. Consignors and carriers of irradiated nuclear fuel, plutonium or high level radioactive wastes wishing to transport these materials by sea are advised of the Code for the Safe Carriage of Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes in Flasks on board Ships (INF Code) to be found in the supplement to the IMDG Code [6]. This code assigns ships carrying these materials to one of three classes depending on the total activity of radioactive material which may be carried and lays down requirements for each class concerning damage stability, fire protection, temperature control of cargo spaces, structural considerations, cargo securing arrangements, electrical supplies, radiological protection equipment and management, training and shipboard emergency plans.

Additional requirements relating to transport by air

576.1. This requirement relates to the presence of passengers on an aircraft rather than its capability to carry passengers. Referring to para. 203, an aircraft equipped to carry passengers, but which is carrying no passengers on that flight, may meet the definition of a cargo aircraft and may be used for the transport of Type B(M) packages and of consignments under exclusive use.

577.1. The special conditions of air transport would result in an increased level of hazard in the case of the types of packages described in para. 577. There may be a considerable reduction in ambient air pressure at the cruising altitudes of aircraft. This is partially compensated for by a pressurization system, but that system is never considered to be 100% reliable.

577.2. If venting were permitted, this would increase considerably as the outside pressure is reduced and it would be difficult to design for this to occur safely. Ancillary cooling and other operational controls would be difficult to ensure within an aircraft under normal and accident conditions.

577.3. Any liquid pyrophoric material poses a special hazard to an aircraft in flight and severe limitations apply to such materials. Where a radioactive substance which has the subsidiary hazard of pyrophoricity is also a liquid, there is a greater probability of a spill occurring, and it is therefore absolutely forbidden to carry such a substance by air transport.

578.1. Because of the higher radiation levels than would normally be allowed, greater care is necessary in loading and handling. The requirement for such consignments to be transported by special arrangement ensures the involvement of the competent authority and allows special handling precautions to be specified, either during loading, in flight or at any intermediate transfer points.

578.2. The special arrangement authorization should include consideration of handling, loading and in-flight arrangements in order to control the radiation doses to flight crew, ground support personnel and incidentally exposed persons. This may necessitate special instructions for crew members, notification to appropriate persons such as terminal staff at the destination and intermediate points and special consideration of transfer to other transport modes.

Additional requirements relating to transport by post

579.1. When shipping by post, special attention should be paid to national postal regulations to ensure that shipments are acceptable to national postal authorities.

579.2. For movement by post, the allowed levels of radioactivity are only one tenth of the levels allowed for excepted packages by other modes of transport, for the following reasons:

(a) The possibility exists of contaminating a large number of letters, etc., which would subsequently be widely distributed, thus increasing the number of persons exposed to the contamination.
(b) This further reduction would result in a concurrent reduction in the maximum radiation level of a source which has lost its shielding, and this is considered to be suitably conservative in the postal environment in comparison with other modes of transport.
(c) A single mailbag might contain a large number of such packages.

580.1. When authorization is given to an organization for the use of the postal service, it would be preferable that one suitably knowledgeable and responsible individual be appointed to ensure that the correct procedures and limitations are observed.

CUSTOMS OPERATIONS

581.1. The fact that a consignment contains radioactive material does not, per se, constitute a reason to exclude such consignments from normal customs operations. However, because of the radiological hazards involved in examining the contents of a package containing radioactive material, it is essential that the examination of the contents of packages be carried out under suitable radiation protection conditions. The presence of a person with adequate knowledge of handling radioactive material and capable of making sound radiation protection judgements is necessary to ensure that the examination is carried out without any undue radiation exposure of customs staff or any third party.

581.2. Transport safety depends, to a large extent, on safety features built into the package. It is thus essential that no customs operation diminish the safety inherent in the package, when the package is to be subsequently forwarded to its destination. Again, the presence of a qualified person will help to ensure the adequacy of the package for its continued transport. `Qualified person' in this context means a person versed in the regulatory requirements for transport as well as in the preparation of the package containing the radioactive material for onward transport.

581.3. For the examination of packages containing radioactive material by customs officials the following advice is given:

(a) Clearance formalities should be carried out as quickly as possible, to eliminate delays in customs clearance which may decrease the usefulness of valuable radioactive material; and

(b) Any necessary internal inspection should be carried out at places where adequate facilities are available and radiation protection precautions can be implemented by qualified persons.

581.4. When it is noted that a package has been damaged, the customs official should immediately contact a qualified person, give them the necessary information and follow the instructions that they issue. No person should either remain near the package (a segregation distance of 3 m would generally be sufficient) or touch it unless absolutely necessary. If handling is necessary some form of protection should be used to avoid direct contact with the package. After handling it is advisable to wash hands.

581.5. When necessary, packages should be placed for temporary storage in an isolated secure place. During such storage, the segregation distance between the packages and all persons should be as great as practicable. Warning signs should be posted around the package and storage area (see also para. 568.1).

UNDELIVERABLE CONSIGNMENTS

582.1. For segregation see para. 568.1

SECTION VI

REQUIREMENTS FOR RADIOACTIVE MATERIALS AND FOR PACKAGINGS AND PACKAGES

REQUIREMENTS FOR RADIOACTIVE MATERIALS

Requirements for LSA-III material

601.1. See para. 226.9.

601.2. The leaching rate limit of 0.1 A_2 per week was arrived at by considering the case of a block of material in its packaging (e.g., a steel drum), which had been exposed to the weather and had taken in sufficient rain for the block to be surrounded with a film of water for one week. If this package is then involved in a handling accident, some of the liquid may escape and, on the basis of the standard model for determining A_2 values, 10^{-4} to 10^{-3} of this may be taken into the body of a bystander (see Appendix I). Since the package must withstand the free drop and stacking tests as prescribed in paras 722 and 723, some credit can be given for its ability to retain some of its contents: it may not be as good as a Type A package but it may well be good enough to limit escape to 10^{-2} to 10^{-3} of the dispersible contents. Since the total body intake must be limited to 10^{-6} A_2 to maintain consistency with the safety built into Type A packages, the dispersible radioactive contents of the drum (i.e., the liquid) must therefore not exceed 0.1 A_2.

Requirements for special form radioactive material

602.1. Special form radioactive material must be of a reasonable size to enable it to be easily salvaged or found after an incident or loss; hence the restriction on minimum size. The figure of 5 mm is arbitrary but is practical and reasonable bearing in mind the type of material normally classified as special form radioactive material.

603.1. The Regulations seek to ensure that a package containing special form radioactive material would not release or disperse its radioactive contents during a severe accident, by leakage from the sealed capsule or by dispersion/leaching of the radioactive material itself, even though the packaging may be destroyed (see Appendix I). This minimizes the predicted hazards from inhalation or ingestion of, or from contamination by, the radioactive material. For this reason special form radioactive material must be able to survive severe mechanical and thermal tests analogous to the tests applied to Type B(U) packages without undue loss or dispersal of radioactive material at any time during its working life.

603.2. The applicant should demonstrate that the solubility of the material evaluated in the leaching test is equal to or greater than that of the actual radioactive material to be transported. Results should also be extrapolated if material with reduced radioactive contents is used in the test, in which case the validity of the extrapolation should be demonstrated. The applicant should not assume that simply because a material is inert that it will pass the leach test without being encapsulated. For example, bare encapsulated iridium 192 pellets have failed the leach test [56]. Leaching values should be scaled up to values reflecting the total activity and form which will be transported. For material enclosed in a sealed capsule suitable volumetric leakage assessment techniques, such as vacuum bubble or helium leakage test methods, may be used. In this case all test parameters which have an effect on sensitivity need to be thoroughly specified and accounted for in evaluating the implied loss of radioactive material from the special form radioactive material.

603.3. The Regulations allow alternative leakage assessment tests for sealed capsules. When, by agreement with the competent authority concerned, the performance tests of a capsule design are not performed with radioactive contents, the leakage assessment may be made by a volumetric leakage method. A rate of 10^{-5} $Pa.m^3/s$ for non-leachable solid contents and a rate of 10^{-7} $Pa.m^3/s$ for leachable solids [56], liquids and gases would in most cases be considered to be equivalent to the activity release of 2 kBq prescribed in para. 603. Four volumetric leak test methods are recommended as being suitable for detecting leaks in sealed capsules and these are listed in Table III together with their sensitivity.

- Leachable: Greater than 0.01% of the total activity in 100 ml in still H_2O at 50°C for 4 hours conforming to 5.1.1 ISO 9978 [57]
- Non - leachable: Less than 0.01% of the total activity in 100 ml in still H_2O at 50°C for 4 hours conforming to 5.1.1 ISO 9978

603.4. When using non-radioactive material as a surrogate, the measurement of leaked material must be related to the limit of activity specified in para. 603(c) of the Regulations.

TABLE III. COMPARISON OF THE FOUR VOLUMETRIC LEAK TEST METHODS RECOMMENDED BY ASTON et al [58]

Leak test method	Sensitivity ($Pa.m^3/s$)	Minimum void in capsule (mm^3)
Vacuum bubble		
(i) Glycol or isopropyl alcohol	10^{-6}	10
(ii) Water	10^{-5}	40
Pressurized bubble with isopropyl alcohol	10^{-8}	10
Liquid nitrogen bubble	10^{-8}	2
Helium pressurization	10^{-8}	10

604.1. Where a sealed capsule constitutes part of the special form radioactive material it is essential to ensure that the capsule offers no possibility of being opened by normal handling or unloading measures. Otherwise the possibility could arise that the radioactive material could be handled or transported without the protecting capsule.

604.2. Sealed sources which can be opened only by destructive techniques are generally assumed to be those of welded construction. They can be opened only by such methods as machining, sawing, drilling or flame cutting. Capsules with threaded end caps or plugs, for example, which may be opened without destroying the capsule, would not be acceptable.

Requirements for low dispersible radioactive material

605.1. Limiting the external radiation level at 3 m from the unshielded low dispersible radioactive material to 10 mSv/h ensures that the potential external dose is consistent with the potential consequences of severe accidents involving Industrial Packages (see para. 521).

605.2. Particles up to about 10 µm aerodynamic equivalent diameter (AED) in size are respirable and can reach deeper regions of the lung, where clearance times may be long. Particles between 10 µm and 100 µm AED are of little concern for the inhalation pathway, but they can contribute to other exposure pathways after deposition. Particles greater than 100 µm AED deposit very quickly. While this could lead to a localized contamination in the immediate vicinity of the accident, it would not represent a significant mechanism for internal exposure.

605.3. For low dispersible material the airborne release of radioactive material in gaseous or particulate form is limited to 100 A_2 when subjecting the contents of a Type B(U) package to the mechanical and thermal tests. This 100 A_2 limit refers to all particle sizes up to 100 µm AED. Airborne releases can lead to radiation exposure of persons in the downwind direction from the location of an aircraft accident via several exposure pathways. Of primary concern is a short term intake of radioactive material through inhalation. Other pathways such as groundshine are much less important because their contribution is only relevant for long resident times and remedial actions can be taken to limit exposure. For the inhalation pathway particles below about 10 µm AED predominate because they are respirable. Nevertheless, a cautiously chosen upper limit of 100 µm was introduced in connection with the 100 A_2 limit. The rationale is that in this way it is assured that neither the inhalation pathway nor other exposure pathways following deposition could lead to unacceptable radiation doses.

605.4. When low dispersible material is subjected to the high velocity impact test particulate matter can be generated but of all airborne particulates up to 100 µm only a small (less than 10%) fraction will be in the respirable size range below 10 µm if the 100 A_2 limit is met. In other words, an equivalent quantity of low dispersible material less than 10 A_2 could be released airborne in a respirable size range. It has been shown that for a reference distance of around 100 m and for a large fraction of atmospheric dispersion conditions this would lead to an effective dose below 50 mSv.

605.5. In the case of the thermal test 100 A_2 of low dispersible material could be released airborne in gaseous form or as particulate with predominantly small (< 10 µm AED) particle sizes because thermal processes such as combustion generally result in small particulates. Attention should be paid to the potential chemical changes of the materials during the enhanced fire test that could lead to aerosol generation, e.g., chemical reactions induced by combustion products. In the case of a fire following an aircraft accident buoyant effects of the hot gases would lead to ground level air concentrations and to potential effective inhalation doses which would also remain below 50 mSv for a large fraction of atmospheric dispersion conditions.

605.6. The limit on leaching of activity is applied to low dispersible radioactive material to eliminate the possibility of dissolution and migration of radioactive material causing significant contamination of land and water courses, even if the low dispersible radioactive material should be completely released from the packaging in a severe accident. The 100 A_2 limit for leaching is the same as that for the release of airborne material consequent to a fire or high velocity impact.

605.7. For the specimen undergoing the impact test, consideration should be given regarding the physical interactions among source structures and individual material components comprising the low dispersible material. These interactions may result in a substantial change of the form of the low dispersible material. For example, a single fuel pellet may not produce the same quantity of dispersible material after a high velocity impact as the same pellet incorporated with other pellets into a fuel rod. It is important that the tested specimen be representative of the low dispersible material that will be transported.

605.8. For the leaching test the specimen should incorporate a representative sample of the low dispersible material, which has been subjected to the enhanced fire test and the high velocity impact test. A separate specimen may be used for each test, in which case two samples would be subjected to the leach test. For example, in the case of the impact test, the material can be broken up or otherwise separated into various solid forms including deposited powder-like material. These forms constitute the low dispersible material that should be subjected to the leaching test.

605.9. It is especially important that the measurements of airborne releases and leached material be reproducible.

GENERAL REQUIREMENTS FOR ALL PACKAGINGS AND PACKAGES

606.1. The design of a package with respect to its securement or retention within or on the conveyance considers only routine conditions of transport, but may also consider the greater demands of normal or even accident conditions of transport.

606.2. For additional guidance on the methods of retaining a package within or on a conveyance, see paras 564.1-564.2 and Appendix V.

607.1. In the selection of materials for lifting attachments, consideration should be given to materials which will not yield under the range of loads expected in normal handling. If overloading occurs, then the safety of the package should not be affected. In addition, the effects of wear should be considered.

607.2. For the design of attachment points of packages lifted many times during their lifetime the fatigue behaviour should be taken into account in order to avoid failure cracks. Where fatigue failure may be assumed, the design should take into account the detectability of those cracks by non-destructive means and appropriate tests should be included in the maintenance programme of the package.

607.3. Snatch factors for lifting should be related to the expected lifting speeds of cranes and be clearly identified. It should be ensured, by an appropriate safety factor against yield, that there is no plastic deformation under these snatch loads in any part of the package.

607.4. Special attention should be given to lifting attachments of packages handled in nuclear facilities. In addition to damage of the package itself, the dropping of heavy mass packages onto sensitive areas could result in radioactive material releases from other sources within the facility or in a criticality or other event which could affect the safety of the facility. For these attachment points even higher safety margins may be required than for normal engineering practice [59],[60],[61].

608.1. This requirement is intended to prevent inadvertent use of package features that are not suitably designed for handling operations.

609.1. This requirement is imposed since protruding features on the exterior of a packaging are vulnerable to impacts during handling and other operations incidental to transport. Such impacts may cause high stresses in the structure of the packaging, resulting in tearing or breaking of containment.

609.2. In determining what is practicable as regards the design and finish of packaging, the primary consideration should be not to detract from the effectiveness of any features which are necessary for compliance with other requirements of the Regulations. For example, features provided for safe handling, operation and stowage should be designed so that while they fulfil their essential functions under the appropriate provisions of the Regulations, any protrusions and potential difficulties of decontamination are minimized.

609.3. Cost is also a legitimate determinant of what is practicable. Measures to comply with para.609 need not involve undue or unreasonable expense. For example, the choice of materials and methods of construction for any given packaging should be guided by commonly accepted good engineering practice for that type of packaging, always having due regard to para. 609, and need not invoke extravagantly expensive measures.

609.4. An exterior surface with a smooth finish having low porosity aids decontamination and is inherently less susceptible to absorption of contamination and subsequent leaching-out ('hide-out') than a rougher one.

610.1. This requirement is imposed because collection and retention of water (from rain or other sources) on the exterior of a package may undermine the integrity of the package as a result of rusting or prolonged soaking. Further, such retained liquid may leach out any surface contamination present and spread it to the environment. Finally, water dripping from the package surfaces, such as rain water, may be misinterpreted as leakage from the package.

610.2. For the purposes of compliance with para. 610 exactly similar considerations to those in paras 609.2 to 609.4 should be applied.

611.1. This requirement is intended to prevent such actions as placing handling tools, auxiliary equipment or spare parts on or near the package in any manner such that the intended functions of packaging components could be impaired both during normal transport and in the event of an accident.

612.1. Components of a packaging, including those associated with the containment system, lifting attachments and retention systems, may be subject to *working loose* as a result of acceleration, vibration, or vibration resonance. Attention should be paid in the package design to ensure that any nuts, bolts and other retention devices remain secure during routine conditions of transport.

613.1. Chemical compatibility of radioactive contents with packaging materials and between different materials of the components of the packagings should take into account such effects as corrosion, embrittlement, accelerated ageing and dissolution of elastomers and elastics, contamination with dissolved material, initiation of polymerization, pyrolysis producing gases, and alterations of a chemical nature.

613.2. Compatibility consideration should include those materials which may be left from manufacturing, cleaning or maintaining the packaging, such as cleaning agents, grease, oil, etc., and also should include residuals of former contents of the package.

613.3. Consideration of physical compatibility should take into account thermal expansion of materials and radioactive contents over the temperature range of concern so as to cover the changes in dimensions, hardness, physical states of materials and radioactive contents.

613.4. One aspect of physical compatibility is observed in the case of liquid contents, where sufficient ullage must be provided in order to avoid hydraulic failure as a consequence of the different expansion rates of the contents and its containment systems within the admissible temperature range. Void volume values to provide sufficient ullage may be derived from regulations for the transport of other dangerous goods with comparable properties.

614.1. Locks are probably one of the best methods of preventing unauthorized operation of valves; they can be used directly to lock the valve closed or used on a lid or cover which prevents access to the valve. Whilst seals can be used to indicate that the valve has not been used they cannot be relied upon to prevent unauthorized operation.

615.1. The materials of the package should be able to withstand changes of ambient pressure and temperature, likely to occur in routine conditions of transport, without impairing the essential safety features of the package.

615.2. An ambient pressure range of 101 kPa to 60 kPa and an ambient temperature range of -40EC to +38EC are generally acceptable for surface modes of transport. For surface movements of excepted package(s), Industrial packages Types IP-1, IP-2 and IP-3, and Type B(M) packages solely within a specified country or solely between specified countries, ambient temperature and pressure conditions other than these may be assumed providing they can be justified and that adequate controls are in place to limit the use of the package(s) to the countries concerned.

ADDITIONAL REQUIREMENTS FOR PACKAGES TRANSPORTED BY AIR

617.1. Surface temperature restrictions are necessary to protect adjacent cargo from potential damage and to protect persons handling packages during loading and unloading. This requirement is particularly restrictive for transport by air as a result of the difficulty of providing adequate free space around packages. For this reason para. 617 always applies to the air mode, whereas for other modes less restrictive surface temperature limits may be applied, under the conditions of exclusive use (see para. 662 of the Regulations and 662.1 to 662.4). If, during transport, the ambient temperature exceeds 38^{o}C under extreme conditions (e.g., see para. 618), the limit on accessible surface temperature no longer applies.

617.2. Account may be taken of barriers or screens intended to give protection to persons without the need for the barriers or screens being subject to any test.

618.1. The ambient temperature range of -40^{o}C to +55^{o}C covers the extremes expected to be encountered during air transport and is the range required by the International Civil Aviation Organization [37] for packaging any dangerous goods, other than "ICAO excepted goods", destined for air transport.

618.2. In designing the containment consideration of the effect of ambient temperature extremes on resultant surface temperatures, contents, thermal stresses and pressure variations is needed to ensure containment of the radioactive material.

619.1. This is a similar provision to that required by the International Civil Aviation Organization [37] for packages containing certain liquid hazardous material intended for transport by air. In this edition of the Regulations the provision has been expanded to include all forms of radioactive material.

619.2. Pressure reductions due to altitude will be encountered during flight (see para. 577.1). The pressure differential which occurs at an increased altitude needs to be taken into account in the packaging design. The 5 kPa is the minimum ambient pressure to be accommodated by the designer (this results from a consideration of aircraft depressurization at a maximum altitude together with a safety margin).

REQUIREMENTS FOR EXCEPTED PACKAGES

620.1. See 515.1

REQUIREMENTS FOR INDUSTRIAL PACKAGES

Requirements for Industrial package Type 1 (Type IP-1)

621.1. According to the radiological grading of LSA material and SCO, the three industrial package types have different safety functions. Whereas Type IP-1 packages simply contain their radioactive contents under routine transport conditions, Type IP-2 packages and Type IP-3 packages protect against loss or dispersal of their contents and loss of shielding under normal conditions of transport, which by definition (see para. 106) include minor mishaps, as far as the test requirements represent these conditions. Type IP-3 packages, in addition, provide the same package integrity as a Type A package intended to carry solids.

621.2. Neither the industrial package design requirements of the Regulations nor United Nations packing group III design requirements regard packages as pressure vessels. In this respect, only those pressure vessels which have a volume of less than 450 l in the case of liquid contents and of less than 1000 l in the case of gaseous contents can be considered as packages. Pressure vessels with greater volumes are defined as tanks, for which paras 625 and 626 provide a comparable level of safety. In the event that pressure vessels are used as industrial packages, the design principles of relevant pressure vessel codes should be taken into account for the selection of materials, design/calculation rules, and quality assurance requirements for the manufacturing and use of the package (e.g., pressure testing by independent inspectors). The comparably high wall thickness of pressure vessels is usually foreseen to provide safety with respect to internal service and/or test pressure. A design pressure higher than that needed to cover service conditions corresponding to the vapour pressure at the upper temperature limit may provide a margin of safety against mishaps or even accidents by necessitating a greater thickness of wall. In this case, it may not be necessary to prove safety by drop- and stacking-performance tests, but rather the pressure test could suffice. However, the safety of associated service equipment (valves etc.) against mechanical loads needs to be assured, for example by the use of additional protective structures.

621.3. Pressure vessels with volumes less than 450 l for liquid contents and 1000 l for gaseous contents, and designed for a pressure of 265 kPa (see para. 625(b)), may provide an adequate level of safety and consequently may not need to be subjected to the Type IP tests. It is understood that all precautions specified by the relevant pressure vessel codes for the use of pressure vessels are taken into consideration and applied as appropriate.

621.4. An example for this application is the pressure vessels used for the transport of uranium hexafluoride (UF_6). These cylinders are designed for a pressure much higher than occurs under normal transport and service conditions. They are therefore inherently protected against mechanical loads.

621.5. The ullage requirement (see para. 647) is not specified as a requirement for the industrial packages. However, in the case of liquid contents, or solid contents such as UF_6 which may become liquid in the event of heating, it is necessary to provide sufficient ullage, as referred to in para. 647, in order to prevent rupture of the containment. Such rupture can occur in the case of insufficient ullage especially as a result of expansion of contents with temperature changes.

Requirements for Industrial package Type 2 (Type IP-2)

622.1. Consideration of the release of contents from Type IP-2 packages impose a containment function on the package for normal conditions of transport. Some simplification in demonstrating no loss or dispersal of contents is possible owing to the rather immobile character of some LSA material and SCO contents and the limited specific activity and surface contamination. See also paras 646.2 to 646.5.

622.2. See paras 621.1 and 226.1

622.3. For a Type IP-2 packaging intended to carry a liquid, see paras 621.2 to 621.5. For a Type IP-2 packaging intended to carry a gas, see paras 621.2 to 621.4. For an Type IP-2 packaging intended to carry LSA-III see para. 226.9.

622.4. For packages exhibiting little external deformation and negligible internal movement of the radioactive contents or shielding, a careful visual examination may provide sufficient assurance that the surface radiation level is essentially unchanged.

622.5. If it is considered that a surface radiation level has probably increased, it is necessary to perform monitoring tests to ensure that the increase in the radiation level does not exceed 20%.

622.6. The method of evaluating the loss of shielding varies from one manufacturer to another. This could lead to discrepancies in evaluating a package's ability to satisfy the requirements of para. 622(b). One way of overcoming this problem may be to define the maximum surface area of the package over which the surface radiation level is assessed. Thus, for example, individual measurements may be taken over areas not greater than 10% of the total surface area of the package. The package surface may be marked to define the subdivisions to be considered and tests conducted by means of a test source suitable for the package (i.e., cobalt-60 or sodium-24 for general package use or specific nuclides for a certain package design). It may be necessary to consider the effect of increased localized radiation levels when evaluating shielding loss. Where other conditions are not specified, the procedures outlined in paras 531.1 to 531.3 may be followed.

622.7. The loss of shielding should be evaluated on the basis of the measurements taken both before and after the tests specified in para. 622 and the resulting data should be compared to determine whether the package satisfies the requirement or not.

Requirements for Industrial package Type 3 (Type IP-3)

623.1. Consideration of the release of contents from Type IP-3 packages impose the same containment function on Type IP-3 packages as for Type A packages for solids, taking into account the higher values of specific activity to be transported in Type IP-3 packages and the absence of operational controls in non-exclusive use transport. In addition, sufficient ullage should be foreseen in the case of liquid LSA material in order to avoid hydraulic failure of the containment system. These requirements are consistent with the graded approach of the Regulations. See also paras 646.2 to 646.5.

623.2. See paras 621.1 and 226.1

623.3. For a Type IP-3 package intended to carry a liquid, see paras 621.2 to 621.5. For a Type IP-3 package intended to carry a gas, see paras 621.2 to 621.4. For a Type IP-3 package intended to carry LSA-III, see para. 226.9.

Alternative requirements for Industrial packages Types 2 and 3 (Type IP-2) and (Type IP-3)

624.1. The alternative use of United Nations packagings is allowed because the United Nations Recommendations [7] require comparable general design requirements and performance tests which have been judged to provide the same level of safety. Whereas leaktightness is also one of the performance test criteria in the United Nations Recommendations, this is not the case with respect to the shielding requirements in the Regulations, which need special attention when United Nations packagings are used.

624.2. As United Nations packing groups I and II require the same or even more stringent performance test standards compared with those for Type IP-2 packages, Type IP-2 test requirements are automatically complied with by all of the United Nations packing groups I and II except as stated in para. 624.3. This means that packagings marked with `X' or `Y' according to the United Nations system are potentially suitable for the transport of LSA and SCO requiring a Type IP-2 package when no specific shielding is required. For these packages, there should be consistency between the contents being shipped and the contents tested in the UN tests, including consideration of maximum relative density, gross mass, maximum total pressure, vapour pressure and the form of the contents.

624.3. United Nations packagings of packing group I and II, i.e., packagings which meet the specifications given in Chapter 9 of the United Nations Recommendations on the Transport of Dangerous Goods [7], may be used as Type IP-2 packages provided there is no loss or dispersal of the contents during or after the UN tests. It should be noted however that a slight discharge from the closure upon impact is permitted under the UN standard if no further leakage occurs. This discharge would not meet the requirement for no loss or dispersal of the contents. In addition the intended contents should be consistent with those allowable in the particular packaging and specific shielding should not be required. The applicable restrictions can be determined from the United Nations marking which must appear on United Nations specification packagings.

625.1. Tank containers designed for the transport of dangerous goods according to international and national regulations have proved to be safe in handling and transport, in some cases even under severe accident conditions.

625.2. The general design criteria for tank containers with respect to safe handling, stacking and transport can be complied with if the structural equipment (frame) is designed in accordance with ISO 1496/3-1990 [62]. This standard prescribes a structural framework in which the tank is attached in such a manner that all static forces of handling, stowage and transport produce no undue stresses on the shell of the tank.

625.3. The dynamic forces under routine conditions of transport are considered in Appendix V.

625.4. Tank containers designed according to ISO1496/3 are considered to be at least equivalent to those that are designed to the standards prescribed in the chapter on Recommendations on Multimodal Tank Transport of the United Nations Recommendations on the Transport of Dangerous Goods.

625.5. The shielding retention requirement (para. 625(c)) is complied with if after the tests the shielding material remains in place, shows no significant cracks and prevents no more than a 20% increase in the radiation level as evaluated by calculation and/or measurements under the above mentioned conditions. In the case of tank containers with an ISO framework, the radiation level calculations/measurements may take the surfaces of the framework as the relevant surfaces.

626.1. To explain the equivalence between tank standards and those prescribed in para. 625 (UN, Chapter 12 for tank-containers) reference should be made to the "European Agreement Concerning the International Carriage of Dangerous Goods by Road (ADR) 1995" [10], where Appendix B.1A prescribes the requirements for road tank vehicles that are basically providing the same safety level as the requirements for tank-containers in Appendix B.1B. A similar comparison can be found in the European Agreement on railway transport [38] (RID) for rail tank-wagons and tank containers at Appendices X and XI of the Agreement.

627.1. Freight containers designed and tested to ISO 1496/1 [4] and approved in accordance with the CSC Convention [5] have been proved, by the use of millions of units, to provide safe handling and transport under routine conditions of transport. It should be noted however that ISO 1496/1 addresses issues relating to container design and testing whereas the CSC Convention is primarily concerned with ensuring that containers are safe for transport, are adequately maintained and are suitable for international shipment by all modes of surface transport. The testing prescribed in CSC is not equivalent to that prescribed in ISO 1496/1.

627.2. Freight containers designed and tested to ISO 1496/1 are restricted to the carriage of solids because they are not regarded as being suitable for free liquids or liquids in non-qualified packagings. Consideration should be given to the construction details of the container to ensure that the containment requirements can be met. Only closed types of freight container can be used to demonstrate compliance with the Type IP-2 and Type IP-3 containment requirement of no loss or dispersal of radioactive contents and monitoring during and after testing is necessary to demonstrate this. Closed types of containers in that sense means also freight containers with openings on top, if these openings are safely closed during transport.

627.3. Freight containers must be shown to retain and contain their contents during accelerations occurring in routine transport because the ISO Standard Tests for freight containers do not include dynamic tests.

627.4. Care must be taken to ensure that attachments used within the container to secure objects can withstand loads typical of routine conditions of transport (see Appendix V).

627.5. For guidance on preventing the loss or dispersal of contents and of the loss of shielding integrity, see paras 622.1 to 622.7.

628.1. Intermediate Bulk Containers approved according to provisions on the basis of Chapter 16 of the United Nations Recommendations on the Transport of Dangerous Goods are considered to be equivalent to packages designed and tested in accordance with the Type IP-2 and Type IP-3 requirements, except with regard to any shielding requirements. The alternative use of Intermediate Bulk Containers is restricted to metal designs only because they provide the closest match with Type IP-2 and Type IP-3 package requirements. The need for other design types could not be identified and they seem not to be appropriate for the transportation of radioactive materials.

628.2. Compliance with the Type IP-2 and Type IP-3 design and performance test requirements may, with the exception of any shielding requirement, be demonstrated for Intermediate Bulk Containers when they conform to provisions based upon the UN Recommendations on the Transport of Dangerous Goods[7], Chapter 16 with the additional requirement for Intermediate Bulk Containers with more than 0.45 m^3 capacity to perform the drop test in the most damaging position (and not only onto the base). These recommendations include comparable design and performance test requirements as well as the design approval by the competent authority.

REQUIREMENTS FOR PACKAGES CONTAINING URANIUM HEXAFLUORIDE

629.1. Uranium hexafluoride is a radioactive material having significant subsidiary hazards. Depending on the degree of enrichment and amount of fissile uranium, uranium hexafluoride may be transported, from the radiological standpoint, in excepted, industrial packages, Type A or Type B packages. Thus, the radiological and fissile properties of uranium hexafluoride are covered by other aspects of the Regulations. However, many of the requirements for uranium hexafluoride imposed by way of ISO 7195 [26], and by the requirements now embodied in the Regulations relate, not to the radiological and fissile hazards posed by uranium hexafluoride, but to the physical properties and also to the chemical toxic hazard of the material when released to the atmosphere and reacted with water or water vapour. In addition, because these packagings are pressurized during loading and unloading operations they have to comply with pressure vessel regulations, although they are not pressurized under normal transport conditions. The requirements specified in paras 629-632 of the Regulations are focused on these non-radiological and fissile concerns. Applicable requirements relating to the radiological and fissile nature of the uranium hexafluoride being packaged and transported, found elsewhere in the Regulations, are vital to providing proper safety during handling and transport, and should therefore be considered in both the packaging and transport of uranium hexafluoride.

630.1. The 0.1 kg exemption level provides assurance against the explosion of small, bare cylinders of UF_6 [63]. The 0.1 kg level is well below the toxic risk limit of 10 kg, based on work by Ringot [64] and Biaggio [65].

630.2. The acceptance criteria in subparas 630(a), 630(b) and 630(c) vary depending upon the type of environment to which the package is exposed. For the pressure test specific to uranium hexafluoride packages (para. 718), the requirement for acceptance without leakage and without unacceptable stress may be satisfied by hydrostatic testing of the cylinder, where leaks may be detected by observing for evidence of water leakage from the cylinder. The valve and other service equipment are not included in this pressure test (ISO 7195).

630.3. For the drop test (para. 722), acceptance may be evidenced by performing a gas leakage test consistent with the procedure, pressure and sensitivity specified for valve leak testing in ISO 7195.

630.4. The criteria for acceptance during or following exposure of a package containing uranium hexafluoride to the thermal test (para. 728) is based upon considerations of the desire to prevent tearing of the cylinder shell. Concerning the allowable release, a necessary acceptance criteria would be the demonstration of "no rupture" of the cylinder, where again consideration is not given to leakage by service equipment such as through and around valves. Consistent with the philosophy used as guidance for "without rupture of the containment system" used in para. 657, tearing or major failure of the uranium hexafluoride cylinder walls would be unacceptable, but minor leakage through or around a valve or other engineered penetration into the cylinder wall may be acceptable subject to competent authority approval.

630.5. It may be difficult if not impossible to demonstrate compliance with the leakage, loss or dispersal, rupture and stress requirements of para. 630 through testing with uranium hexafluoride in the packagings because of major environmental, health and safety concerns. Thus, demonstration of compliance may need to depend upon surrogates for the uranium hexafluoride in tests combined with reference to previous satisfactory demonstrations, laboratory tests, calculations and reasoned arguments as elaborated upon in para. 701.

630.6. For the demonstration of compliance of packages containing uranium hexafluoride with the requirements of para. 630(c), the designer should take into account the influence of the parameters that may alter the transient thermo-physical conditions of uranium hexafluoride and the packaging which may be encountered in the thermal test. The designer should consider, at a minimum, the following:

(a) The most severe orientation of the package. Changing the orientation of the package might produce a different distribution of the three physical phases of uranium hexafluoride (solid, liquid and gas) inside the package, and could lead to different consequences on internal pressure [66],[67]
(b) The full range of allowed filling ratios. The pressure inside the cylinder could be dependent, in a complex fashion, upon the extent to which it is filled. For example, for very small filling ratios, the solid uranium hexafluoride could melt and evaporate faster, thereby accelerating the pressure increase inside the package [68].
(c) The actual properties of the structural materials at high temperatures. For instance, a high reduction in tensile strength of steel occurs at temperatures above 500°C [69].
(d) The presence of metallurgical defects in the structure material could cause the rupture of the package. This would be a function of the defect size. The maximum design defect size should be derived from design analyses, the manufacturing process, and inspection acceptance criteria.
(e) Thinning of the wall of the cylinder or other packaging components resulting from corrosion could result in reduced performance. The designer should establish a minimum acceptable wall thickness and methods for determining wall thicknesses for both unfilled and filled, in-service cylinders should be developed and applied [70],[71].

631.1. This provision is included since it is unlikely that a pressure relief device can be provided which is sufficiently reliable to assure a desired level of release and subsequent closure once the pressure reduces to acceptable levels. However, should a design of a pressure relief valve for venting an overpressurized, overheated cylinder of uranium hexafluoride become available which can be demonstrated to function properly and reliably, then use of this may be allowed subject to approval of the competent authority as specified in para. 632(a).

632.1. Packages designed to carry 0.1 kg or more of uranium hexafluoride which are not designed to withstand the 2.8 MPa pressure test, but are designed to withstand a pressure test of at least 1.4 MPa may be authorized for use subject to approval by the competent authority. This is to allow older package designs which can be demonstrated to the satisfaction of the competent authority to be safe, to be used subject to multilateral approval. The package designer should prepare the safety case for justifying this certification.

632.2. Very large packages containing uranium hexafluoride, designed to contain 9,000 kg or more of uranium hexafluoride, which are not transported in thermal protecting overpacks have been considered to possibly have sufficient thermal mass to survive without rupture of the containment system exposure to the thermal test of para. 728. Subject to approval of the competent authority, these packages may be certified for shipment on a multi-lateral basis, and the package designer should prepare the safety case for justifying this certification.

632.3. See also 630.5.

REQUIREMENTS FOR TYPE A PACKAGES

634.1. The minimum dimension of 10 cm has been adopted for a number of reasons. A very small package could be mislaid or slipped into a pocket. In order to conform to international transport practice package labels have to be 10 cm square. To adequately display these labels the dimensions of the packages are required to be at least 10 cm.

635.1. Requiring a package seal is intended both to discourage tampering and to ensure that the recipient of the package knows whether or not the contents and/or the internal packaging have been tampered with or removed during transport. While the seal remains intact the recipient is assured that the contents are those stated on the label; if the seal is damaged, the recipient will be warned that extra caution will be required during handling and particularly on opening the package.

635.2. The type and mass of the package will, in the main, dictate the type of security seal to be used, but designers should ensure that the method chosen is such that it will not be impaired during normal handling of the package in transport.

635.3. There are many methods of sealing but the following are typical of those used on packages for radioactive material:

(a) When the packaging is a fibreboard carton, gummed or self-adhesive tape which cannot be re-used to seal the package may be used (the outer packaging and/or the tape will be effectively destroyed on being opened).

(b) Crimped metal seals may be used on the closures of drums, lead and steel pots and small boxes. The seals are crimped onto the ends of a suitable lace or locking wire and are embossed with an identifying pattern. The method used to secure the closure itself should be independent of the security seal.

(c) Padlocks may be used on timber boxes and also for steel or lead/steel packages. A feature such as a drilled pillar is incorporated into the box or packaging design so that when the padlock is fitted through the drilled hole it is not possible to gain entry into the package.

636.1. With the exception of tanks or packages used as freight containers, the tie-down to the vehicle will in general be a standard piece of equipment suitable for restraining packages which have a considerable variation in mass. It may therefore be sensible to design the attachment of the tie-down to the package as the weak link; this can be done by designing the attachment point so that it will only accommodate a certain maximum size of shackle pin, be held by pins that would shear, or bolts that would break, at a designated stress.

636.2. Lifting points may be used as tie-down attachments, but if so used they should be designed specifically for both tasks. The separate lifting points and tie-down attachments should be clearly marked to indicate their specific purposes, unless they can be so designed that alternative use is impossible, e.g., a hook type of lifting point cannot normally be used for tie-down purposes.

636.3. Consideration can also be given to potential directional failure of the tie-down systems so the transport workers are protected in the event of head-on impacts, while the package is protected against excessive side loads from side-on impacts [72]. For details on recommended design considerations of packages and their retention systems, see Appendix V.

637.1. Type A package components should be designed for a temperature range from -40EC to + 70EC corresponding to possible ambient temperatures within a vehicle or other enclosure or package temperatures when the package is exposed to direct sunlight. This range covers the conditions likely to be encountered in routine conditions of transport and storage in transit. If a wider environmental temperature range is likely to be encountered during transport or handling or there is significant internal heat generation, then this should be allowed for in the design. Some of the items that may need consideration are:

- expansion/contraction of components relative to structural or sealing functions;
- decomposition or changes of state of component materials at extreme conditions;
- tensile/ductile properties and package strength; and
- shielding design.

638.1. There are many existing national and international standards covering an extremely wide range of design influences and manufacturing techniques, e.g., pressure vessel codes, welding standards, leaktightness standards, etc., which can be used in the design, manufacturing and testing of packages. Designers and manufacturers should, wherever possible, work to these established standards in order to promote and demonstrate adequate control in the overall design and manufacture of packages. The use of such standards also means that the design and manufacture processes are more readily understood by all relevant people, sometimes in different locations and Member States, involved in the various phases of transport; most importantly package integrity is much less likely to be compromised.

638.2. Where new or novel design, manufacturing or testing techniques are proposed for use and there is no appropriate existing standard, the designer may need to discuss the proposals with the Competent Authority to obtain acceptance. Consideration should be given by the designer, Competent Authority, or other responsible bodies to developing an acceptable standard covering any new design concept, manufacturing or testing technique, or material to be used.

639.1. Examples of positive fastening devices which may be suitable are:

- welded seams
- screw threads
- snap-fit lids
- crimping
- rolling
- peening
- heat shrunk materials, and
- adhesive tapes or glues.

Other methods may be appropriate depending on the package design.

640.1. In the case of packages where containment of the radioactive contents is achieved by means of special form radioactive material, attention is drawn to the requirements of para. 502(f) with respect to each shipment.

642.1. Certain materials may react chemically or radiolytically with some of the substances intended to be carried in Type A packages. Tests may be required to determine the suitability of materials to ensure that the containment system is neither susceptible to deterioration caused by the reactions themselves, nor damaged by the pressure increase consequent upon those reactions.

643.1. This requirement is intended to prevent an excessive pressure differential arising in a package that has been filled at sea level (or below) and is then carried by surface transport to a higher altitude. The minimum requirement for packages subject to air pressure variations resulting from altitude changes is that resulting from surface movements to altitudes as high as 4000 metres. If the package could be sealed at or below sea level and transported over land to this altitude, the package must be able to withstand an overpressure resulting from this change in altitude as well as being able to withstand any overpressure that may be generated by its contents.

643.2. For guidance on the requirement for the retention of radioactive contents, see paras. 646.2 to 646.6.

644.1. To prevent contamination caused by leakage of contents through valves, a provision for some secondary device or enclosure for these valves is required by the Regulations. Depending upon the specific design, such a device or enclosure may help to prevent the unauthorized operation of the valve, or in such an event prevent the contents from escaping.

644.2. Examples of enclosures which may be suitable are:

(a) Blank caps on threaded valves using gaskets;
(b) Blank flanges on flanged valves using gaskets; and
(c) Specially designed valve covers or enclosures, using gaskets, designed to retain any leakage.

Other methods may be appropriate depending on the package design.

645.1. The requirement of this paragraph is primarily intended to ensure that the radiation shield is constantly maintained around the radioactive substance to minimize any increase in radiation levels on the surface of the package. When the radiation shield is a separate unit the positive fastening device ensures that the containment system is not released except by deliberate intent.

645.2. Examples of design features which may be suitable are:

(a) Hinge operated interlock devices on covers;
(b) Bolted, welded or padlocked frames surrounding the radiation shield; and
(c) Threaded shielding plugs.

Other methods may be appropriate depending on the package design.

646.1. The design of, and contents limits imposed upon, Type A packages intrinsically limit any possible radiologicalhazard. This paragraph provides the restrictions on release and degradation of shielding during normal conditions of transport so as to ensure safety.

646.2. A maximum allowable leakage rate for the normal transport of Type A packages has never been defined quantitatively in the Regulations but it has always been required in a practical sense.

646.3. Practically, it is difficult to advise on a single test method that could satisfactorily incorporate the vast array of packagings and their contents that exist. A qualitative approach, dependent upon the packaging under consideration and its radioactive contents, may be employed. In applying the preferred test method the maximum differential pressure used should be that resulting from the contents and the expected ambient conditions.

646.4. For solid, granular and liquid contents, one way of satisfying the requirements for `no loss or dispersal' would be to monitor the package (containing a non-active, control material) on completion of a vacuum test or other appropriate tests to determine visually whether any of the contents have escaped. For liquids, an absorbent material may be used as a test indicator. Thereafter, a careful visual inspection of the package may confirm that its integrity is maintained and no leakage has occurred. Another method which may be suitable in some cases would be to weigh the package before and after a vacuum test to determine whether any leakage has occurred.

646.5. For gaseous contents, visual monitoring is unlikely to be satisfactory and a suction detection or pressurization method with a readily identifiable gas (or volatile liquid providing a gaseous presence) may be used. Again, a careful visual inspection of the packaging may confirm that its integrity has been maintained and no escape paths exist. Another detection method would be a simple bubble test.

646.6. For advice concerning loss of shielding integrity, see paras. 622.4 to 622.7.

647.1. Ullage is the gas-filled space available within the package to accommodate the expansion of the liquid contents of the package due to changes in environmental and transport conditions. Adequate ullage ensures that the containment system is not subjected to excessive pressure due to the expansion of liquid-only systems, which are generally regarded as incompressible.

647.2. When designing ullage requirements it may be necessary to consider both extremes of package material temperature, -40EC and +70EC (see para. 637). At the lower temperature, pressure increases may occur as a result of expansion at transitional temperatures where the material changes its state from liquid to solid. At the higher temperature, pressure increases may occur as a result of expansion or vaporization of the liquid contents. Consideration may also be needed to ensure that excessive ullage is not provided as this may allow unacceptable dynamic surges within the package during transport. In addition, surging or lapping may occur during filling operations involving large liquid quantities and designers may need to consider this aspect for certain package designs.

648.1. The purpose of these two additional requirements is to demonstrate either an increased capability of a Type A packaging for liquids to withstand impacts and hence to indicate that the fraction of the contents that would be released in an accident would be comparable to that released from a Type A package designed to carry dispersible solids, or to provide a supplementary safety barrier, thereby reducing the probability of the liquid escaping from the package even if it escapes from the primary inner containment components.

648.2. A user of a Type B(U) package, or a Type B(M) package, may wish to use that package for shipping less than an A_2 quantity of liquid and to designate this package in the shipping papers as a Type A package shipment. This lifts some administrative burdens from the consignor and carrier, and since the package has a greater integrity than a standard Type A package, safety is not degraded. In this case, it is not required to meet the provision of adding absorbent material or a secondary outer containment component.

649.1. The reasons for additional tests for Type A packaging for compressed or uncompressed gases are similar to those for Type A packagings for liquids (see para. 648.1). However, since in the case of gases, failure of the containment would always give 100% release, the additional test is required to reduce the probability of failure of the containment for a given severity of accident and thus achieve comparable risk to that of a Type A package designed to carry dispersible solids.

649.2. The exception of packages containing tritium or noble gases from the requirement in para. 649 is based upon the dosimetric models for these materials (the Q system, see discussion in Appendix I).

649.3. For guidance on the requirement of no loss or dispersal of gaseous radioactive contents, see para. 646.5.

REQUIREMENTS FOR TYPE B(U) PACKAGES

650.1. The concept of a Type B(U) package is that it is capable of withstanding most of the severe accident conditions in transport without loss of containment or increase in external radiation level to an extent which would endanger the general public or those involved in rescue or clean up operations. It could be safely recovered (see paras. 510 and 511), but it would not necessarily be capable of being reused.

651.1. Although the requirement in para. 637, which is for Type A packages, is intended to cover most conditions which can result in packaging failure, additional consideration of packaging component temperatures is required for Type B(U) packages on a design specific basis. This is generally because Type B(U) packages may be designed for contents which produce significant amounts of heat, and component temperatures for such a design may exceed the 70EC requirement for Type A packages. The intent of specifying an ambient temperature of 38EC for package design considerations is to ensure that the designer properly addresses packaging component temperatures and the effect of these temperatures on geometry, shielding, efficiency, corrosion and surface temperature. Furthermore, the requirement that a package be capable of being left unattended for a period of one week under an ambient temperature of 38EC with solar heating is intended to ensure that the package will be at, or close to, equilibrium conditions and that under these conditions it will be capable of withstanding the normal transport conditions, demonstrated by tests according to paras. 719 - 724, without loss of containment or reduction in radiation shielding.

651.2. The evaluation to ambient temperature conditions must account for heat generated by the contents, which may be such that the maximum temperature of some package components may be considerably in excess of the maximum of 70EC required for a Type A package design.

651.3. See also paras 637.2, 652.1 and 652.2, 654.1 to .9 and 664.1 to .3 and Appendix VI.

651.4. Practical tests may be used to determine the internal and external temperatures of the package under normal conditions by simulating the heat source due to radioactive decay of the contents with electrical heaters. In this way, the heat source can be controlled and measured. Such tests should be performed in a uniform and steady thermal environment (i.e., fairly constant ambient temperature, still air, and minimum heat input from external sources such as sunlight). The package with its heat source should be held under test for sufficient time to allow the temperatures of interest to reach steady state. The test ambient temperature and internal heat source should be measured and used to adjust linearly all measured package temperatures to those corresponding to a 38EC ambient temperature.

651.5. For tests performed in uncontrolled environments (e.g., outside) ambient variations (e.g., diurnal) may make it impossible to achieve constant steady state temperatures. In such cases, the periodic quasi-steady state temperatures should be measured (both ambient and package) allowing correlations to be made between ambient and package average temperatures. These results, together with data on the internal heat source, can be used to predict package temperatures corresponding to a steady 38EC ambient temperature.

652.1. The surface temperatures of packages containing heat generating radioactive materials will rise above the ambient temperature. Surface temperature restrictions are necessary to protect adjacent cargo from potential damage and to protect persons handling packages during loading and unloading.

652.2. With a surface temperature limit of 50°C at the maximum ambient temperature of 38°C other cargo will not become overheated nor will anyone handling or touching the surface suffer a burn. A higher surface temperature is permitted under exclusive use (except for transport by air), see para. 662 and paras. 662.1 to 662.4.

653.1. See para. 664.1.

654.1. During transport, a package could be subjected to solar heating. The effect of solar heating is to increase the package temperature. To avoid the difficulties in trying to account for the many variables precisely, values for insolation have been agreed upon internationally (they are presented in Table XI of the Regulations). The insolation values are specified as uniform heat fluxes applied for 12 hours and followed by 12 hours of zero insolation. Packages are assumed to be in the open; therefore, neither shading nor reflection from adjacent structures is considered. Table XI shows a maximum value of insolation for an upward facing horizontal surface and zero for a downward facing horizontal surface which receives no insolation. A vertical surface is assumed to be heated only half a day and only half as effectively; therefore, the table value for insolation of a vertical surface is given as one quarter the maximum value for an upward facing flat surface. Locations on curved surfaces vary in orientation between horizontal and vertical, and are judiciously assigned half the maximum value for upward facing horizontal surfaces. The use of the agreed upon values ensures uniformity in any safety assessment, providing a common ground for the purpose of calculation.

654.2. The insolation data provided in Table XI of the Regulations are uniform heat fluxes. They are to be applied at the levels stated for 12 hours (daylight) followed by 12 hours of no insolation (night). The cyclic step functions representing insolation should be applied until the temperatures of interest reach conditions of steady periodic behaviour.

654.3. A simple but conservative approach for evaluating the effects of insolation is to apply uniform heat flux continuously at the values stated in Table XI. Use of this approach avoids the need to perform transient thermal analysis; only a simple steady state analysis is performed.

654.4. For a more precise model, a time dependent sinusoidal heat flux may be used to represent insolation during daylight hours for flat surfaces or for curved surfaces. The integrated (total) heat input to a surface between sunrise and sunset is required to be equal to the appropriate value of total heat for the table values over 12 hours (i.e., multiply the table value by 12 hours to get total heat input in W/m^2). The period between sunset and sunrise gives zero heat flux for this model. The cyclic insolation model should be applied until the temperatures of interest reach conditions of steady periodic behaviour.

654.5. Downward facing flat surfaces cannot receive any insolation, and the Table XI value of `none' applies. For any upward facing horizontal surface, the Table XI value is applicable. Non-horizontal surfaces may include vertical or nearly vertical surfaces (i.e., up to 15E off the vertical); for these surfaces, the Table XI value for vertical surfaces applies. For upward tilted flat surfaces that are more than 15E off the vertical, the horizontal projection of the area may be used in conjunction with the insolation value for a flat upward facing horizontal surface. For downward tilted flat surfaces that are more than 15E off the vertical, the vertical projection of the area may be used in conjunction with the insolation value for a flat vertical surface.

654.6. The insolation value for curved surfaces given in Table XI should be applied to all curved surfaces of any orientation.

654.7. Components of the package that reduce insolation to any surface (i.e., provide solar shade to the surface of the package) may be taken into account in the thermal evaluation. Any such components assumed to reduce insolation should not be included in the thermal evaluation if their effectiveness would be reduced as a result of the package being subjected to the tests for normal conditions of transport.

654.8. Because radiation heat transfer depends on the emissivity and absorptivity at a surface, variations in these properties may be taken into account. These surface properties are wavelength dependent. Solar radiation corresponds to high temperature and short wavelength radiation while surface radiation from packages corresponds to relatively low temperature and longer wavelength radiation. In many cases, the absorptivity will be lower than the emissivity, so using the higher value for both will give a larger margin of safety when the objective is heat dissipation. In other cases, advantage might be taken of naturally occurring differences in these properties, or the surface could be treated to take advantage of such differences to reduce the effect of insolation. When differences in surface properties are used as a means of thermal protection to reduce insolation effects, the performance of the thermal protection system should be demonstrated, and the system should be shown to remain intact under normal conditions of transport.

654.9. Evaluation of the package temperature for transport of radioactive material may be done by analysis or test. Tests if used should be performed on full scale models. If the radiation source is not sunlight, differences between solar wavelength and the source wavelength should be taken into account. The test should continue until thermal equilibrium is achieved (either constant steady state or steady periodic state, depending on the source). Corrections should be made for ambient temperatures and internal heat, where necessary.

655.1. In general, coatings for thermal protection fall into two groups, those which undergo a chemical change in the presence of heat (e.g., ablative and intumescent) and those which provide a fixed insulation barrier (including ceramic materials).

655.2. Both groups are susceptible to mechanical damage. Materials of the ablative and intumescent type are soft and can be damaged by sliding against rough surfaces (such as concrete or gravel) or by the movement of hard objects against them. In contrast, ceramic materials are very hard, but are usually brittle and unable to absorb shock without cracking or fracturing.

655.3. Commonly occurring incidents, which could cause damage to the thermal protection materials include: relative movement between package and contact surfaces of vehicle during transport; skidding across a road in which surface gravel is embedded; sliding over a damaged rail track or against the edge of a metal member; lifting or lowering against bolt heads of adjacent structures or equipment; impact of other packages (not necessarily containing radioactive material) during stowage or transport; and many other situations which would not result from the tests required in paras 722 to 727. Packages that are tested by a simple drop test do not receive damage to the surface representative of the rolling and sliding action usually associated with a vehicle accident and packages subsequently thermally tested may have a coating which under practical accident conditions could be damaged.

655.4. The damage to a thermal protection coating may reduce the effectiveness of the coating, at least over part of the surface. The package designer should assess the effects of this kind of damage.

655.5. The effects of age and environmental conditions on the protective material also need to be taken into account. The properties of some materials change with time, and with temperature, humidity or other conditions.

655.6. A coating may be protected by adding skids or buffers which would prevent sliding or rubbing against the material. A durable outer skin of metal or an overpack may give good protection but could alter the thermal performance of the package. The external surface of the package may also be designed so that thermal protection can be applied within recesses.

655.7. With the agreement of the competent authority, thermal tests with arbitrary damage to the thermal protection of a package may be made, to show the effectiveness of damaged thermal protection, where it can be shown that such damage will yield conservative test results.

656.1. The concept of specifying containment standards for large radioactive source packages in terms of activity loss in relation to specified test conditions was first introduced in the 1967 Edition of the Regulations.

656.2. The release rate limit of not more than $A_2 \times 10^{-6}$ per hour for Type B(U) packages following tests to demonstrate their ability to withstand the normal conditions of transport was originally derived from considerations of the most adverse expected condition. This was taken to correspond to a worker exposed to activity leaking from a package during its transport by road in an enclosed vehicle. The design principle embodied in the Regulations is that intentional release of activity from a Type B(U) package should be avoided. However, since absolute containment cannot be guaranteed, the purpose of specifying maximum allowable activity leak rates is to permit the specification of appropriate and practical test procedures which are related to acceptable radiological protection criteria. The model used in the derivation of the release rate of $A_2 \times 10^{-6}$ per hour is that exposure of workers is highly unlikely to exceed the annual dose and intake limits for radiation workers set forth in the Basic Safety Standards (BSS). In addition, it is shown that alternative exposure scenarios involving workers exposed in a store room or cargo handling area, or persons exposed out of doors, are less restrictive.

656.3. The 1973 Revised Edition (As Amended) of the Regulations stipulated that the radiation level at 1 m from the surface of a Type B(U) package should not exceed 100 times the value that existed before the accident condition tests, had the package contained a specified radionuclide. This requirement constituted an unrealistic design constraint in the case of packages designed to carry other radionuclides. Therefore, since the 1985 Edition of the Regulations, a specific maximum radiation level of 10 mSv/h has been stipulated, irrespective of radionuclide.

656.4. The release limits of not more than 10 A_2 for krypton-85 and not more than A_2 for all other radionuclides in a period of one week for Type B(U) packages when subjected to the tests to simulate normal and accident conditions of transport represent a simplification of the provisions of the 1973 Edition of the Regulations. This change was introduced in recognition of the fact that the Type B(U) limit appeared unduly restrictive in comparison with safety standards commonly applied at power reactor sites [73],[74], especially for severe accident conditions which are expected to occur only very infrequently. The radiological implications of a release of A_2 from a Type B(U) package under accident conditions have been discussed in detail elsewhere [75]. Assuming that accidents of the severity simulated in the Type B(U) tests specified in the Regulations would result in conditions such that all persons in the immediate vicinity of the damaged package would be rapidly evacuated, or be working under health physics supervision and control, the incidental exposure of persons otherwise present near the scene of the accident is unlikely to exceed the annual dose or intake limits for workers set forth in the BSS. The special provision in the case of krypton-85, which is the only rare gas isotope of practical importance in shipments of irradiated nuclear fuel, results from specific consideration of the dosimetric consequences of exposure to a radioactive plume, for which the models used in the derivation of A_2 values for non-gaseous radionuclides are inappropriate [76].

656.5. The Regulations require Type B(U) packages to be designed to restrict loss of radioactive contents to an acceptable low level. This is specified as a permitted release of radioactivity expressed as a fraction of A_2 per unit time for normal and accident conditions of transport. These criteria have the advantage of expressing the desired containment performance in terms of the parameter of primary interest: the potential hazard of the particular radionuclide in the package. The disadvantage of this method is that direct measurement is generally impractical and it is required to be applied to each individual radionuclide in question in the physical and chemical form which is expected after the mechanical, thermal and water immersion tests. It is more practical to use well established leakage testing methods such as gas leakage tests, see ANSI N14.5 [31] and ISO 12807 [32]. In general leakage tests measure material flow passing a containment boundary. The flow may contain a tracer material such as a gas, liquid, powder or the actual or surrogate contents. A means should therefore be determined to correlate the measured flow with the radioactive material leakage expected under the reference conditions. This radioactive material leakage can then be compared with the maximum radioactive material leakage rate that is permitted by the Regulations. If the tracer material is a gas, the leakage rate expressed as a mass flow rate can be determined. If the tracer material is a liquid, either the leakage rate, expressed as a volumetric flow rate, or the total leakage expressed as a volume can be determined. If the tracer material is a powder, the total leakage, expressed as a mass, can be determined. Finally if the tracer material is radioactive, the leakage expressed as an activity can be determined. Volumetric flow rates for liquids and mass flow rates for gases can be calculated by the use of established equations. If powder leakage is calculated by assuming that the powder behaves as a liquid or an aerosol, the result will be very conservative.

656.6. The basic calculative method therefore involves the knowledge of two parameters, the radioactive concentration of the contents of the package and its volumetric leak rate . The product of these two parameters should be less than the maximum permitted radioactivity leakage rate expressed as a fraction of A_2 per unit time.

656.7. For packages containing radioactive materials in liquid or gaseous form the concentration of the radioactivity is to be determined in order to convert Bq/h (activity leak rate) to m^3/s (volumetric leak rate) under equivalent transport conditions. When the contents include mixtures of radionuclides (R1, R2, R3 etc.) the `unity rule' specified in para. 404 is used as follows:

$$\frac{\textit{ase of R1}}{\textit{ease of R1}} \;\%\; \frac{\textit{Potential release of R2}}{\textit{Allowable release of R2}} \;\%\; \frac{\textit{Potential relec}}{\textit{Allowable rele}}$$

656.8. From this, and assuming uniform leakage rates over the time intervals being considered, the radioactivity of the gas or liquid in the package and the volumetric leakage rate are required to fulfil the following:

For the conditions in para. 656(a)

$$\mathbf{j}_{i} \; \frac{C_{(Ri)}}{A_{2(Ri)}} \; \# \; \frac{10^{\&6}}{3600L} \;\; ' \;\; \frac{2.78 \times 10^{\&10}}{L}$$

For the conditions in para. 656(b)(ii)

$$\sum_i \frac{C_{(Ri)}}{A_{2(Ri)}} \le \frac{1}{7 \times 24 \times 3600 L} = \frac{1.65 \times 10^{-6}}{L}$$

where,

$C_{(Ri)}$ is the concentration of each nuclide in TBq/m^3 of liquid or gas at standard conditions of temperature and pressure (STP);
$A_{2(Ri)}$ is the limit specified in Table I in TBq for that nuclide;
L is the permitted leak rate in m^3/s of liquid or gas at STP.

The quantity C can also be derived as follows:

$$C = GS$$

where,

G is the concentration of the nuclide in kg/m^3 of liquid or gas at STP;
S is the specific activity of the nuclide in TBq/kg of the pure nuclide (see Appendix II) or

$$C = F\overline{G}S$$

where,

F is the fraction of the nuclide present in an element (percentage/100);
G is the concentration of the element in kg/m^3 of liquid or gas at STP.

656.9. Note that the allowable activity release after tests for normal conditions of transport is given in terms of TBq/h and after tests for accident conditions in terms of TBq/week. It is unlikely that any leakage after an accident will be at a uniform rate. The value of interest is the total leakage during the week and not the rate at any time during the week (i.e., the package may leak at a high rate for a short period of time following exposure to the accident environment and then release essentially nothing for the remainder of the week as long as the total release does not exceed A_2 per week).

656.10. The calculated permitted radioactive liquid or gas leakage may then be converted to an equivalent test gas leakage under reference conditions, taking account of pressure, temperature and viscosity by means of the equations for laminar and/or molecular flow conditions, examples of which are given in American National Standard ANSI N14.5-1977 [31] or ISO [DIS] 12807 [32]. In particular cases where a high differential pressure may result in a high permitted gas velocity, turbulent flow may be the more limiting and should be taken into account.

656.11. The test gas leakage determined by the above method may range from about 1 Pa m^3/s to less than 10^{-10} Pa m^3/s, depending upon the A_2 values of the radionuclides and their concentration in the package. Generally in practice, a test need not be more sensitive than 10^{-8} Pa m^3/s for a pressure difference of 1×10^5 Pa to qualify a package as being leaktight. Where the estimated allowable test leakage rate exceeds 10^{-2} Pa m^3/s, a limiting value of 10^{-2} Pa m^3/s is recommended because it is readily achievable in practical cases.

656.12. When a package is designed to carry solid particulate material, test data on the transmission of solids through discrete leak paths or seals can be used to establish test gas conditions. This will generally give a higher allowed volumetric leak rate than by assuming the particulate material behaves as a liquid or an aerosol. In practice even the smallest particle size powder would not be expected to leak through a seal which has been tested with helium to better than 10^{-6} Pa m^3/s with a pressure difference of 1×10^5 Pa.

656.13. In a package design, maximum radiation levels are established both at the surfaces (para. 532) and at 1 m from the surfaces of the package (as implied by para. 532 coupled with para. 526). After the tests for accident conditions have been performed, however, an increase in the radiation level is allowed provided that the limit of 10 mSv/h at 1 m from the surface is not exceeded when the package is loaded with its maximum allowed activity.

656.14. When shielding is required for a Type B(U) package design, the shielding may consist of a variety of materials, some of which may be lost during the tests for accident conditions. This is acceptable provided that the radioactive contents remain in the package and sufficient shielding is retained to ensure that the radiation level at 1 m from the `new' (after test) external surface of the package does not exceed 10 mSv/h.

656.15. The demonstration of compliance with this acceptance criterion of not more than 10 mSv/h at 1 m from the external surface of a Type B(U) package after the applicable tests may be made by different means: calculations, tests on models, parts or components of the package, tests on prototypes, etc., or by a combination of them. In verifying compliance, attention should be paid to the potential for increased localized radiation levels emanating through cracks or gaps which could appear as a defect of design or manufacturing or could occur during the tests as a consequence of the mechanical or thermal stresses, particularly in drains, vents and lids.

656.16. When the verification of compliance is based on full scale testing, the evaluation of the loss of shielding may be made by putting a suitable radioactive source into the specimen and monitoring entirely the outside surface with an appropriate detector, for instance films, Geiger-Müller probes or scintillation probes. For thick shields a scintillation probe, e.g., NaI (thallium activated), of small diameter (about 50 mm) is usually employed because it allows the use of low activity sources, typically cobalt-60, and because its high sensitivity and small effective diameter permits an easy and effective detection of increased localized radiation levels. If measurements are made near the surface of the packaging, care must be taken to properly measure (see para. 531.1) the radiation level and to average the results (see para. 531.2). Calculations will then be needed to adjust the measured radiation level to 1 m from the external surface of the package. Finally, unless the radioactive contents for which the package is designed are used in the test, further calculations will be required to adjust the measured values to those which would have existed had the design contents been used.

656.17. The use of lead as a shielding material needs special care. It has a low melting temperature and high coefficient of expansion and, therefore, it should be protected from the effects of the thermal test. If it is contained in relatively thin steel cladding which could be breached in the impact test and if the lead melts in the fire it would escape from the package. Also, owing to its high coefficient of expansion the lead could burst the cladding in the thermal test and be lost. In both these cases the radiation level could be excessive after the thermal test. To overcome the expansion problem voids might be left to allow the lead to expand into them but it should be recognized that, when the lead cools, a void will exist whose position may be difficult to predict. A further problem is that uniform melting of the lead may not necessarily occur, owing to non-uniformities in packaging structure and in the fire environment. In this event, localized expansion could result in the cladding being breached and the subsequent loss of lead, thus reducing the shielding capability of the package.

656.18. Additional guidance on testing the integrity of radiation shielding may be found in the literature [28],[29],[77],[78],[79],[80].

656.19. Packages designed for the transport of irradiated fuel pose a particular problem in that the radioactivity is concentrated as fission products in fuel pins which have been sealed prior to irradiation. Pins which were intact on loading into the package would generally be expected to retain this activity under normal conditions of transport.

656.20. Under accident conditions of transport irradiated fuel pins may fail with subsequent release of activity into the package containment system. Data on the fuel fission product inventory, possible failure rate of pin cladding and the mechanism of activity transfer from the failed pin into the containment system are therefore required to enable the package leaktightness to be assessed.

656.21. The above methods of assessing the leaktightness requirements of packages are generally applied in two ways:

(a) When the package is designed for a specific function, the radioactive contents are clearly defined and the standard of leaktightness can be established at the design stage.
(b) When an existing package with a known standard of leaktightness is required to be used for a purpose other than that for which it was designed, the maximum allowable radioactive material contents has to be determined.

656.22. In the case of a mixture of radionuclides leaking from a Type B(U) package, an effective A_2 may be calculated by the method of para. 404, using the fractional activities of the constituent radionuclides f(i) which are appropriate to the form of mixture which can actually leak through the seals. This is not necessarily the fraction within the package itself since part of the contents may be in solid discrete pieces too large to pass through seal gaps. In general, for leakage of liquids and gases the fractional quantities relate to the gaseous or dissolved radionuclides. Care is necessary, however, to take account of finely divided suspended solid material.

656.23. If the package has elastomeric seals, permeation of gases or vapours may cause relatively high leakage rates. Permeation is the passage of a liquid or gas through a solid barrier (which has no direct leak paths) by an absorption-diffusion process. Where the radioactive material is gaseous (e.g., fission gas) the rate of permeation leakage is determined by the partial pressure of the gas and not by the pressure in the containment system. The tendency of elastomeric materials to absorb gases can also be taken into account.

656.24. It should be noted that, in the case of some large packages, very small leakage of activity over a long time period could result in contamination of the exterior surface. In these cases it may be necessary to reduce the leakage under normal conditions of transport (para. 656(a)) to ensure that the surface contamination limit (paras 214, 508 and 509) is not exceeded.

657.1. Various risk assessments have been carried out over the years for the sea transport of radioactive materials, including those documented in the literature [81],[82]. These studies considered the possibility of a ship carrying packages of radioactive material sinking at various locations; the accident scenarios included a collision followed by sinking, or a collision followed by a fire and then followed by sinking.

657.2. In general it was found that most situations would lead to negligible harm to the environment and minimal radiation exposure to persons if the packages were not recovered following the accident. It was found, however, that should a large irradiated fuel package (or packages) be lost on the continental shelf, some long term exposure to persons through the ocean food-chain could occur. The radiological impact from loss of irradiated fuel packages at greater depths or of other radioactive material packages at any depth was found to be orders of magnitude lower than these values. Later studies have considered the radiological impact from the loss of other radioactive materials which are increasingly being transported in large quantity by sea, such as plutonium and high-level radioactive waste. Based on these studies, the scope of the enhanced water immersion test requirement has been extended in the 1996 Edition of these Regulations to cover any radioactive material transported in large quantity, not only irradiated nuclear fuel.

657.3. In the interest of keeping the radiological impacts as low as reasonably achievable should such an accident occur, the requirement for a 200 m water submersion test for irradiated fuel packages containing more than 37 PBq of activity was originally added to the 1985 Edition of the Regulations. In this edition of the Regulations the threshold defining "large quantity" has been amended to a multiple of A_2, which is considered a more appropriate criterion to cover all radioactive materials, being based on a consideration of external and internal radiation exposure to persons as a result of an accident. The 200 m depth corresponds approximately to the continental shelf and to the depths where the above mentioned studies indicated radiological impacts could be important. Recovery of a package from this depth would be possible and often would be desirable. Although the influence of radioactivity release into the environment would be acceptable, as shown by the risk assessments, the requirement in para. 657 was imposed because salvage would be facilitated after the accident if the containment system were not ruptured, and therefore only retention of solid contents in the package was considered necessary. Under such circumstances the gross retention of the containment result in a containment system which is completely unimpaired by immersion in 200 m of water. The specific release rate requirements imposed for other test conditions (see para. 656) are therefore not applied here.

657.4. In many cases of Type B(U) package design, the need to meet other sections of the Regulations will result in a containment system which is completely unimpaired by immersion in 200m of water.

657.5. In cases in which the containment efficiency is impaired, it is recognized that leakage into the package and subsequent leakage from the package is possible.

657.6. The aim under conditions of an impaired containment should be to ensure that only dissolved activity is released. Retention of solid radioactive material in the package reduces the problems in salvaging the package.

657.7. Degradation of the total containment system could occur with prolonged immersion and the recommendations made in the above paragraphs should be considered as being applicable, conservatively, for immersion periods of about 1 year, during which recovery should readily be completed.

658.1. The increase in design complexity and any additional uncertainty and possible unreliability associated with filters and mechanical cooling systems are not consistent with the philosophy underlying the Type B(U) designation (unilateral competent authority approval). The simpler design approach where neither filters nor cooling systems are used has a much wider acceptability.

660.1. Subsequent to the closure of a package the internal pressure may rise. There are several mechanisms which could contribute to such a rise including exposure of the package to a high ambient temperature, exposure to solar heating (i.e., insolation), heat from the radioactive decay of the contents, chemical reaction of the contents, radiolysis in the case of water filled designs, or combinations thereof. The maximum value which the summation of all such potential pressure contributors can be expected to produce under normal operating conditions is referred to as the maximum normal operating pressure (MNOP) - see paras. 228.1 to 228.4.

660.2. The presence of such a pressure could adversely affect the performance of the package and consequently needs to be taken into account in the assessment of performance under normal operating conditions.

660.3. Similarly, in the assessment of the ability to withstand accident conditions (paras 726-729) the presence of a pre-existing pressure could present more onerous conditions against which satisfactory package performance must be demonstrated -- consequently the MNOP needs to be assumed in defining the pre-test condition. (See paras 228.1 and 228.2). If justifiable, pressures different from the MNOP may be used provided the results are corrected to reflect the MNOP.

660.4. Type B(U) packages are generally not pressure vessels and do not fit tidily within the various codes and regulations which cover such vessels. For the tests required to verify the ability of a Type B(U) package to withstand both normal and accident conditions of transport, assessment under the condition of maximum normal operating pressure (MNOP) is required. Under normal transport conditions the prime design considerations are to provide adequate shielding and to restrict the leakage of radioactivity under quite modest internal pressures. The accident situation represents a single extreme incident following which re-use is not considered as a design objective. Such an extreme incident is characterized by single short duration, high stress cycles during the mechanical tests at normal operating temperature, followed by a single, long duration stress cycle induced by the temperatures and pressures created during the thermal test. Neither of these stressing cycles fit the typical pattern of loading of pressure vessels, the design of which is concerned with time dependent degradation processes such as creep, fatigue, crack growth and corrosion. For that reason, specific reference to the allowable stress levels has not been included in the Regulations. Instead, strains in the containment system are restricted to values which will not affect its ability to meet the applicable requirements. Whilst other requirements might eventually assume importance, it is for the containment of radioactivity that the containment system exists. Before a fracture would occur it is likely that containment systems, particularly in re-usable packagings with mechanically sealed joints, will leak. It is of prime importance therefore to determine the extent to which the strains in the various components distort the containment system and impair its sealing integrity. Reduction of seal compression brought about, for example, by bolt extensions and local damage due to impact and by rotations of seal faces during thermal transients need to be assessed. One assessment technique is to predict the distortions on impact directly from drop tests on representative scale models and to combine these with the distortions calculated to arise during the thermal test using a recognized and validated computer code. The effects upon sealing integrity of the total distortion may then be determined by experiments on representative sealed joints with appropriately reduced seal compressions.

660.5. The MNOP should be determined in accordance with the definition given in para. 228.

660.6. It is recommended that the strains in a containment system under normal conditions of transport at maximum normal operating pressure should be within the elastic range. The strains under accident conditions of transport should not exceed the strains which would allow leakage rates greater than those stated in para. 656(b), nor increase the external surface radiation level beyond the requirements of para.

656.

660.7. When analysis is used to evaluate package performance, the MNOP should be used as a boundary condition for the calculation of the effect of the tests for demonstrating ability to withstand normal conditions of transport and as an initial condition for the calculation of the effect of the tests for demonstrating ability to withstand accident conditions of transport.

661.1. The requirement that the MNOP not exceed 700 kPa gauge is the specified limit for Type B(U) packages which is considered to be generally acceptable to all competent authorities.

662.1. The surface temperature limit of 85EC for Type B(U) packages under exclusive use, where potential damage to adjacent cargo can be well controlled, is required to prevent injury to persons from casual contact with packages. When exclusive use does not apply, or for all air transport, the surface temperature is limited to 50EC to avoid potential heat damage to adjacent cargo. The barriers or screens referred to in para. 662 are not regarded as part of the package design from the standpoint of radiological safety; therefore, they are excluded from any tests associated with package design.

662.2. Insolation may be ignored with regard to the temperature of accessible surfaces and account is taken only of the internal heat load. The justification for this simplification is that any package, with or without internal heat, would experience a similar surface temperature increase when subjected to insolation.

662.3. Readily accessible surface is not a precise description, but is interpreted here to mean those surfaces which could be casually contacted by a person who may not be associated with the transport operation. For example, the use of a ladder might make surfaces accessible, but this would not be cause for considering the surfaces as readily accessible. In the same sense, surfaces between closely spaced fins would not be regarded as readily accessible. If fins are widely spaced, say the width of a person's hand or more, then the surface between the fins could be regarded as readily accessible.

662.4. Barriers or screens may be used to give protection against higher surface temperatures and still retain the Type B(U) approval category. An example would be a closely finned package fitted with lifting trunnions where the use of the trunnions would require the fins to be cut away locally to the trunnions and thus expose the main body of the package as an accessible surface. Protection may be achieved by the use of a barrier, such as an expanded metal screen or an enclosure which effectively prevents access or contact with the package by persons during routine transport. Such forms of protection would then be considered as accessible surfaces and would thus be subject to the applicable temperature limit. The use of barriers or screens should not impair the ability of the package to meet heat transfer requirements nor reduce its safety. Such a screen or other device is not required to survive the regulatory tests for the package design to be approved. This provision permits approval of packages using such thermal barriers without the barriers having to be subjected to the tests which the package is required to withstand.

663.1. Special attention should be given to the interaction between the low dispersible radioactive material and the packaging during normal and accident conditions of transport. This interaction should not damage the encapsulation, cladding, or other matrix nor cause comminution of the material itself to a degree that would change the characteristics as demonstrated by the requirements of para. 605.

664.1. The lower temperature is important because of pressure increases from materials which expand upon freezing (e.g., water), because of possible brittle fracture of many metals (including some steels) at reduced temperature and because of possible loss of resilience of seal materials. Of these effects, only fracture of materials could lead to irreversible damage. Some elastomers which provide good high temperature performance (e.g., fluorocarbons, such as Viton compounds) lose their resilience at temperatures of -20^{0}C or less. This can lead to narrow gaps of some µm width arising from differential thermal expansion between the metal components and the elastomer. This effect is fully reversible. In addition, freezing of any humid contents and internal pressure drop at the low temperatures could prevent leakage from the containment. Therefore in certain cases the use of such elastomeric seals could be accepted. See references [83],[84] for further information. The lower temperature limit of -40EC and the upper temperature limit of 38EC are reasonable bounding values for ambient temperatures which could be experienced during transport of radioactive material in most geographical regions at most times of the year. However, it must be recognized that in certain areas of the world (extreme northern and southern regions during their winter periods and dry desert regions during their summer periods) temperature extremes below -40EC and above 38EC are possible. Averaged over area and time, however, temperatures falling outside the range -40EC to +38EC are expected to occur during only a small fraction of the time.

664.2. See Appendix VI for Guidelines for Safe Design of Shipping Packages against Brittle Fracture.
664.3. In assessing a package design for low temperature performance, the heating effect of the radioactive contents (which could prevent the package component temperature from falling to the ambient of -40EC) should be ignored. This will allow package response (including structural and sealing material behaviour) at the low temperature to be evaluated for handling, transport and in-transit storage conditions. Conversely, in evaluating a package design for high temperature performance, the effect of the maximum possible heating by the radioactive contents, as well as insolation and the high temperature of 38EC, should be considered simultaneously.

REQUIREMENTS FOR TYPE B(M) PACKAGES

665.1. The intent is that the safety standards of Type B(M) packages, so designed and operated, provide a level of safety equivalent to that provided by Type B(U) packages.

665.2. Departures from the requirements given in paras 637, 653, 654 and 657 to 664 are acceptable, in some situations, with the agreement of the pertinent competent authority(s). An example of this could be a reduction in the ambient temperature range and insolation values taken for design purposes if the Type B(U) requirements are not considered applicable (paras 637, 653, 654 and 664), or making allowance for the heating effect of the radioactive contents.

666.1. For the contents of some packages, as a result of the mechanisms described in para. 660.1, the pressure tends to build up and if not relieved might eventually cause failure of the package, or reduce the useful lifetime of the package through fatigue. To avoid this, para. 666 allows the package design to include a provision for intermittent venting. Such vented packages are required by the Regulations to be shipped as Type B(M) packages.

666.2. In order to provide safety equivalent to that which would be provided by a Type B(U) package, the design may include requirements that only gaseous materials should be allowed to be vented, that filters or alternative containment might be used, or that venting may only be performed under the direction of a qualified health physicist.

666.3. Intermittent venting is permitted in order to allow a package to be relieved of a buildup of pressure which might, under normal conditions of transport (see paras 719-724) or when the package is subjected to the thermal test (see para. 728), cause it to fail to meet the Regulations. Activity release under normal conditions and under accident conditions, where no operational controls are used, is limited, however, by the provisions of para. 656.

666.4. Because there is no specified regulatory limit of activity release for intermittent venting, where operational controls are used the person responsible should be able to demonstrate to the competent authority, using a model which relates as closely as possible to the actual conditions of package venting, that transport workers and members of the public will not be exposed to doses in excess of those laid down by the relevant national authorities. When the intermittent venting operation is taking place under the control of a radiation protection adviser, the release may be varied on his advice taking account of measurements made during the operation to assure that workers and members of the public are adequately protected.

666.5. Factors taken into account in such an assessment will include:

(a) Exposure due to normal activity leakage and external radiation from the package;
(b) The location and orientation of the venting orifice in relation to the working position of the operator and the proximity of workers and members of the public;
(c) Occupancy factors of workers and members of the public;
(d) The physical and chemical nature of the material being vented, e.g., gaseous (halogen, inert gas, etc.), particulate, soluble/insoluble; and
(e) Other dose commitments incurred by operators and the public.

666.6. In assessing the adequacy of the release operation, account should be taken of possible detriment from retaining and disposing of the released activity rather than allowing it to disperse.

REQUIREMENTS FOR TYPE C PACKAGES

667.1. Analogous to a Type B(U) or Type B(M) package, the concept of a Type C package is that it is capable of withstanding severe accident conditions in air transport without loss of containment or increase in external radiation level to an extent that would endanger the general public or those involved in rescue or cleanup operations. The package could be safely recovered, but it would not necessarily be capable of being reused.

668.1. One of the potential post-crash environments is package burial. Packages involved in a high-velocity crash may be covered by debris or buried in soil. If packages whose contents generate heat become buried, an increase in package temperature and internal pressure may result.

668.2. To make this analysis, the initial condition of the package is taken as it is designed to be presented for transport.

668.3. Demonstration of compliance with the performance standards under burial conditions should be made using conservative calculations or validated computer codes. The evaluation of the condition of a buried package should take into account the integrity of both the thermal insulation shielding and the containment system, according to the requirements specified in para. 669(b). For this reason special attention should be given to heat dissipation capability and the change in the internal pressure in the burial condition.

669.1. The Type C package provides similar levels of protection for the surface and air modes when compared to a Type B(U) or Type B(M) package in a severe accident. To achieve this goal, it is necessary to ensure that the same external radiation level and loss of contents limits are required following the Type B accident condition and the Type C tests.

669.2. See also para. 656 for further explanatory material on requirements for dose limits and material release limits.

669.3. The text in para. 656.1 to 656.24 also applies to Type C packages.

670.1. Because a Type C package may be immersed in a lake, inland sea, or on the continental shelf where recovery is possible, the enhanced immersion test is required for all Type C packages regardless of the total activity in the package.

670.2. In an air accident over a body of water, a package could be submerged for a period of time pending recovery. Large hydrostatic pressures could be applied to the package, depending upon the depth of submersion. Of primary concern is the possible rupture of the containment system. An additional consideration is recovery of the package before severe corrosion develops.

670.3. The 200 m depth required corresponds approximately to the maximum depth of the continental shelf. Recovery of a package from this depth would be possible and desirable. The acceptance criteria for the immersion test is that there is no rupture of the containment system. Further advice may be found in 657.2, 657.3 and 657.5 - 657.8.

670.4. As the sea represents a softer impact surface than land, it is sufficient that the immersion test be an individual demonstration requirement; that is, non-sequential to other tests.

REQUIREMENTS FOR PACKAGES CONTAINING FISSILE MATERIAL

671.1. The requirements for packages containing fissile material are additional requirements imposed to ensure that packages with fissile material contents will remain subcritical under normal and accident conditions of transport. All other relevant requirements of the Regulations must be adhered to. The system for implementing criticality control in transport is prescribed in Section V of the Regulations.

671.2. Packages containing fissile material have to be designed and transported in such a way that an accidental criticality is avoided. Criticality is achieved when the fission chain reactions become self supported due to the balance between the neutron production and the neutron loss by absorption in and leakage from the system. Package design involves consideration of many parameters that influence neutron interaction (see Appendix VII). The criticality safety assessment must consider these various parameters and ensure that the system will remain subcritical in both normal and accident conditions of transport. Assessments should be performed by qualified persons experienced in the physics of criticality safety.

In addition to the obvious control of fissile material mass, the package designer may influence criticality control by any of the following methods.

(i) Selection of the shapes for the confinement system or packaging influence neutron leakage from fissile units by altering the surface-to-volume ratio. For example, thin cylinders or slabs have increased neutron leakage in comparison with spheres or cylinders with a height-to-diameter ratio near unity.

(ii) Selection of packaging material influences the number of leaking neutrons that are reflected back into the fissile material. The number of neutrons returned (or leaving) and their energies are determined to a large extent by the selection of the packaging material.

(iii) Selection of external package dimensions. Neutrons leaking from a package containing fissile material may enter other fissile packages and produce a fission event. Neutron interaction can be influenced by the package dimensions, which determine the spacing of the fissile material and can be adjusted to limit interaction between different units of fissile material.

(iv) Use of fixed neutron absorbers to remove neutrons (see para. 501.8).

(v) Selection of package design to control the ratio of moderating material-to-fissile material ratio, including the reduction of void space to limit the amount of water that could leak into a package.

671.3. The contingencies required to be considered in the assessment of a package presented for shipment, as itemized in subpara. (a), could influence the neutron multiplication of the package or array of packages. These contingencies are typical ones that may be important and should be carefully considered in the assessments. However, depending on the package design and any special conditions anticipated in transport or handling, other atypical contingencies may need to be considered to ensure that subcriticality is maintained under all credible transport conditions. For example, if the test results show movement of the fissile or neutron absorber material in the package, the uncertainty limits that bound this movement should be considered in the criticality safety assessments. It is important to bear in mind that the prototype used in testing may vary from the production models in detail, in manufacturing method and in manufacturing quality. The as built dimensions of the prototype may need to be known to examine the effect of tolerances on the tests. The difference between tested models and production models needs to be considered. The goal is to obtain the maximum credible neutron multiplication such that subcriticality is assured.

671.4. Water influences criticality safety in several ways. When it is mixed with fissile material the resulting neutron moderation can significantly reduce the amount of fissile material required to achieve criticality. As a reflector of neutrons, water also increases the neutron multiplication factor, though less dramatically. If the water reflector is located outside the confinement system it is less effective, and less still outside the package. Thick layers of full-density water (approx. 30 cm) between packages can reduce neutron interaction in arrays to an insignificant value [85],[86]. The criticality assessment should consider the changes in package geometry or conditions that might cause water to behave more as a moderator than a reflector, or vice versa. All forms of water should be considered including: snow, ice, steam, vapour and sprays. These low density forms of water often produce (particularly in considering interstitial water between packages) a neutron multiplication higher than that seen with full density water (see Appendix VII).

671.5. Neutron absorbers are sometimes employed in the packaging to reduce the effect of moderation and the contribution to the neutron multiplication resulting from interaction among packages (see 501.8). Typical neutron-absorbing materials used for criticality control are most effective when a neutron moderator is present to reduce the neutron energy. The loss of effectiveness of neutron absorbers, e.g., by corrosion and redistribution, or as in the case of contained powders, by settling, can have a marked effect on the neutron multiplication factor.

671.6. Subparas (a)(iii) and (iv) address contingencies arising from dimensional changes or movement of the contents during transport. The credible rearrangement of the packaging or contents must be considered in establishing the margin of subcriticality. Changes to the package dimensions due to the normal or accident tests must be of concern to the package evaluator. Indications of dimension changes during the accident tests should cause the evaluator to assess the sensitivity of these changes to the neutron multiplication. A loss of the fissile material from the array of packages considered in the evaluation of para. 682 must be limited to a subcritical quantity. This subcritical quantity should be consistent with the type of contents and with optimum water moderation and reflection by 20 cm of full-density water. The reduction of spaces between packages, credible because of possible damage to the package in transport, will have a direct effect on the neutron interaction among packages; thus, it requires examination. The effect on reactivity of tolerances on dimensions and material compositions should be considered. It is not always obvious whether particular dimensions or compositions should be maximized or minimized or how, in combination, they affect reactivity. A number of calculations may need to be performed in order that the maximum reactivity of the system can be determined or an appropriate allowance for these contingencies can be developed.

671.7. The effects of temperature changes (subpara. (a)(vi)) on the stability of fissile material form or on the neutron interaction properties are to be examined. For example, uranium systems dominated by very low-energy (thermal) neutrons have an increase in neutron multiplication as the temperature is reduced. Temperature changes may also influence the package integrity. The temperatures which should be considered include those resulting from ambient condition requirements specified in para. 676 and those of the tests (paras 728 or 736, as appropriate).

Exceptions from the requirements for packages containing fissile material

672.1. Packages containing fissile material which meet any of the requirements in subparas (a)-(d) are excepted from the criticality safety assessment specified in para. 671(b). Assurance that the excepted criteria is met for both the individual package and the consignment is the responsibility of the consignor of the excepted material.

672.2. The origin of the limits in subpara. (a)(i) is based on the work of Woodcock and Paxton[87] where a minimum container volume of one litre and a maximum limit of 250 packages were used to obtain fissile material limits of 9.4 g for Pu-239, 16.0 g for U-233 and 16.2 g for U-235 for individual packages. Practical considerations (consistency and the fact that the A_2 value for Pu-239 would cause gram quantities to be transported as special form radioactive material or in a Type B packaging) caused the limit to be subsequently changed [88] to a uniform value of 15 g. In subpara. (a)(ii) the minimum critical concentration for Pu-239 is 7.5g/1, and approximately 12 g/1 for U-235 and U-233 for water moderated systems [89]. These values correspond, respectively, to fissile-to-hydrogen mass ratios of approximately 6.7% and 10.8%. Thus, hydrogenous mixtures with less than a 5% fissile-to-hydrogen mass ratio has an adequate subcritical safety margin. Although use of a mass ratio in the exception criteria may be more cumbersome than a concentration value (as used in previous Editions of the Regulations) this formulation is a better measure for hydrogenous mixtures other than water.

Subpara. (a)(iii) facilitates the safe transport of contaminated waste containing fissile material at a very low concentration.

The safety considerations underlying the three exceptions in subpara. (a) are based upon the assumption of hydrogenous moderation and reflection, thus a restriction on the presence of the potentially more effective elements beryllium and deuterium is applied.

Each of the exceptions provided by subpara. (a) are further restricted by an allowed mass limit per consignment. The formula for the mass limit allows for mixing of fissile material, but the formula and the values provided in Table XII are set such that the maximum consignment mass is no more than approximately half a critical mass value. Thus, the exception criteria provide two points of control (individual package and consignment) to prevent the accumulation of fissile material into quantities that might lead to potential criticality.

672.3. The 1% enriched U-235 limit of subpara. (b) is a rounded off value slightly lower than the minimum critical U-235 enrichment for infinite homogenous mixtures of uranium and water published by Paxton and Pruvost [89]. The homogeneity addressed in subpara. (b) is intended to preclude latticing of slightly enriched uranium in a moderating medium. There is agreement that homogeneous mixtures and slurries are those in which the particles in the mixture are uniformly distributed and have a diameter no larger than 127 μm [90],[91] i.e., not capable of passing through a 120 mesh screen. Concentrations can also vary throughout the material, however variations in concentration of the order of 5% should not compromise criticality safety.

672.4. The exception limit for subpara. (c) provides for uranyl nitrate solution to have a content enriched in U-235 to no more than 2% by mass of uranium. This limit is slightly lower than the minimum critical enrichment value reported by Paxton and Pruvost [89].

672.5. Subpara. (d) sets a 1 kg limit for shipments of plutonium containing no more than 20% by weight of plutonium-239 and plutonium-241. Subcriticality in the transport of this quantity of plutonium is assured by virtue of the Type B(U) or Type B(M) packages, which provide adequate separation from other fissile material, and because the plutonium composition is not amenable to criticality in thermal fissioning systems. (Monte Carlo analyses indicate 6.8 kg of material with 80% Pu-238 and 20% Pu-239 by weight is needed for the critical mass of a fully water reflected metal sphere) [92].

672.6. The exceptions provided in para. 672 were originally conceived to ensure that incredible conditions would have to occur for the excepted packages on a conveyance to cause a criticality accident. Besides the accumulation of fissile material mass on a conveyance, the material would have to be subsequently re-arranged within an appropriate moderating material to obtain the density and form required for a critical system. Where necessary the exceptions provide limits on the consignment to preclude the accumulation of critical mass. Shippers and competent authorities should be alert to potential abuses of the exception limits that might give rise to a potential for criticality.

672.7. Other data to support the exceptions limits provided in para. 672 can be found in the literature [92],[93],[94],[95].

Content specification for assessments of packages containing fissile material

673.1. Values of unknown or uncertain parameters should be appropriately selected to produce the maximum neutron multiplication factor for the assessments performed in paras 671-682. In practice this requirement may be met by covering the effect of these uncertainties by a suitable allowance in the acceptance criteria. Mixtures whose contents are not well-defined are often generated as by-products of production operations, e.g., contaminated work clothes, gloves, or tools, residues of chemical analyses and operations, floor sweepings, etc. and as direct products from waste-processing operations. It is important to determine the combination of parameters that produce the maximum neutron multiplication. Thus, the criticality safety assessment must both identify the unknown parameters and explain the interelationship of the parameters and their effects on neutron multiplication. One should determine the range of values possible (based on available information and consistent with the nature of the material involved) for each parameter and show that the neutron multiplication factor for any possible combination of parameter values will satisfy the acceptance criteria. This principle should apply to the irradiation characteristics used to determine the isotopics for irradiated nuclear fuel.

674.1. The requirements for the criticality assessment of irradiated nuclear fuel are addressed in this paragraph. The major objective is to assure that the isotopic contents used in the safety assessment provide a conservative estimate of the neutron multiplication in comparison with the actual loading in the package. Irradiation of fissile material typically depletes the fissile nuclide content and produces actinides, which contribute to neutron production and absorption, and fission products which contribute to neutron absorption. The long-term, combined effect of this change in the nuclide composition is to reduce the reactivity from that of the unirradiated state. However, reactor fuel designs that incorporate fixed neutron burnable poisons can experience an increase in reactivity for short-term irradiations where the reactivity gain due to depletion of the fixed neutron poisons is greater than the reactivity loss due to the change in the fuel composition. If the assessment uses an isotopic composition that does not correspond to a condition greater than or equal to the maximum neutron multiplication during the irradiation history, then the assumed composition of the fissile material should be demonstrated to provide a conservative neutron multiplication for the known characteristics of the irradiated nuclear fuel as loaded in the package.

674.2. Unless it can be demonstrated in the criticality assessment that the maximum neutron multiplication during the credible irradiation history is provided, a pre-shipment measurement must be performed in order to assure the fissile material characteristics meet the criteria (e.g., total exposure and decay) specified in the assessment (see para. 502.8). The requirement for a pre-shipment measurement is consistent with the requirement to assure the presence of fixed neutron poisons (see para. 501.8) or removable neutron poisons (see para. 502.4), where required by the certificate, that are used for criticality control. In the case of irradiated nuclear fuel, the depletion of the fissile nuclides and the buildup of neutron-absorbing actinides and fission products are providing a criticality control that must be assured.

674.3. The maximum neutron multiplication often occurs in the unirradiated state. However, one method of extending the useful residence time of fissile material in a reactor is to add a distributed, fixed neutron burnable poison, allowing a larger initial fissile nuclide content than would otherwise be present. These reactor fuel designs with burnable poisons can experience an increase in reactivity for short-term irradiations where the reactivity gain due to depletion of the fixed neutron poisons is greater than the reactivity loss due to the change in the fuel composition. No pre-shipment measurement is required when such fuel is treated in the criticality assessment as both unirradiated and unpoisoned since this will provide a conservative estimate of the maximum neutron multiplication during the irradiation history. The requirements of para. 674a apply, therefore, not those of 674b. In addition, breeder reactor fuel and production reactor fuel may have multiplication factors that could increase with irradiation time.

674.4. The evaluation of the neutron multiplication factor for irradiated nuclear fuel must consider the same performance standards as required for unirradiated nuclear fuel (see paras 677-682). However, the assessment for irradiated nuclear fuel must determine the isotopic composition and distribution consistent with the information available on the irradiation history. The isotopic composition of a particular fuel assembly in a reactor is dependent, to varying degrees, on the initial isotopic abundance, the specific power, the reactor operating history (including moderator temperature, soluble boron, and reactor assembly location etc.), the presence of burnable poisons or control rods, and the cooling time after discharge. Seldom, if ever, are all of the irradiation parameters known to the safety analyst. Therefore, the requirements of para. 673 regarding unknown parameters must be considered. The information typically available for irradiated nuclear fuel characterization is the initial fuel composition, the average assembly burnup and the cooling time. Data on the operating history, axial burnup distribution, and presence of burnable poisons must typically be based on general knowledge of reactor performance for the irradiated nuclear fuel under consideration. It must be demonstrated that the isotopic composition and distribution determined using the known and assumed irradiation parameters and decay time will provide a conservative estimate of the neutron multiplication factor after taking into account biases and uncertainties. Conservatism could be demonstrated by ignoring all or portions of the fission products and/or actinide absorbers or assuming lower burnup than actual. The axial isotopic distribution of an irradiated fuel assembly is very important because the region of reduced burnup at the ends of an assembly may cause an increased reactivity in comparison to an assembly where the average burnup is assumed for the isotopics over the entire axial height[96],[97],[98].

674.5. Calculational methods used to determine the neutron multiplication should be validated, preferably against applicable measured data (see Appendix VII). For irradiated nuclear fuel this validation should include comparison with measured isotopic data. The results of this validation should be included in determining the uncertainties and biases normally associated with the calculated neutron multiplication. Fission product cross sections can be important in criticality safety analyses for irradiated nuclear fuel. Fission product cross sections measurements and evaluations over broad energy ranges have not been emphasized to the extent that actinide cross sections have. Therefore, the adequacy of fission product cross sections used in the assessment needs to be considered and justified by the safety analyst.

Geometry and temperature requirements

675.1. This requirement applies to the criticality assessment of packages in normal conditions of transport. The prevention of entry of a 10 cm cube was originally of concern when open, 'bird-cage' types of packages were permitted. This requirement can now be viewed as providing a criterion for evaluating the integrity of the outer container of the package. Packages exist which have similar features to the birdcage design but whose protrusions beyond the closed envelope (the bird) of the packaging exist not to provide spacing between units in an array, but, for example, as impact limiters. Where no credit is taken for these features in the spacing of units, a 10 cm cube behind or between the protrusions but outside the closed envelope of the packaging should not be considered to have “entered” the package.

676.1. Departure from the temperature range of -40°C to 38°C is acceptable in some situations, with the agreement of the competent authority. Where the assessment of the fissile aspects of the package in relation to its response to the regulatory tests would be adversely affected by ambient temperatures, the competent authority should specify in the certificate of approval the ambient temperature range for which the package is approved.

Assessment of an individual package in isolation

677.1. Because of the significant effect water can have on the neutron multiplication of fissile materials, the criticality assessment of a package requires the consideration of water being present in all void spaces within a package to the extent causing maximum neutron multiplication. The presence of water may be excepted from those void spaces protected by special features that must remain watertight under accident conditions of transport. Credible conditions of transport that might provide preferential flooding of packages leading to an increase in neutron multiplication should be considered.

677.2. To be considered “watertight” for the purposes of preventing inleakage or outleakage of water related to criticality safety, the effect of both the normal and accident condition tests need to be considered. Definitive leakage criteria for "watertighness" should be set in the safety assessment report (SAR) for each package, and accepted by the competent authority. These criteria should be demonstrated to be achieved in the tests, and achievable in the production models.

677.3. The neutron multiplication for packages containing uranium hexafluoride is very sensitive to the amount of hydrogen in the package. Because of this sensitivity, careful attention has been given to restrict the possibility of water leaking into the package. The persons responsible for testing, preparation, maintenance, and transport of these packages should be aware of the sensitivity of the neutron multiplication in uranium hexafluoride to even small amounts of water and assure that the special features defined here are strictly adhered to.

678.1. The portion of the package and contents that make up the confinement system (see para. 209.1) must be carefully considered to assure that the system includes the portion of the package that maintains the fissile material configuration. Water is specified as the reflector material in the regulations because of its relatively good reflective properties and its natural abundance. The specification of 20 cm of water reflection is selected as a practical value (an additional 10 cm of water reflection would add less than 0.5% in reactivity to an infinite slab of ^{235}U) that is very near the worst reflection conditions typically found in transport. The assessment should consider the confinement system reflected by 20 cm of full density water and with the confinement system reflected by the surrounding material of the packaging. The situation that yields the highest neutron multiplication should be used as the basis for assuring subcriticality. The reason that both situations must be considered is that it is possible that during routine loading operations, or subsequent to an accident, the confinement system could be outside the packaging and reflected by water.

679.1. The requirements for demonstrating subcriticality of an individual package are specified so as to determine the maximum neutron multiplication in both normal and accident conditions of transport. In the assessment, due account must be given to the results of the package tests required in subparas 681(b) and 682(b) and the conditions under which the absence of water leakage may be assumed as described in para. 677.

679.2. Note that “subcritical” means that the maximum neutron multiplication, adjusted appropriately by including a calculational bias, uncertainties, and a subcritical margin, should be less than 1.0. See Appendix VII for specific advice on the assessment procedure and advice on determining an upper subcritical limit.

680.1. It is possible for accidents in the air mode to be significantly more severe than in the surface mode. In recognition of this, more stringent requirements have been introduced in the 1996 Edition of the Regulations for packages designed for the air transport of fissile material.

680.2. The requirements for packages transported by air address separate aspects of the assessment and apply only to the criticality assessment of an individual package in isolation. Subpara. (a) requires a single package, with no water inleakage, to be subcritical following the Type C test requirements of para. 734. This requirement is provided to preclude a rapid approach to criticality that may arise from potential geometrical changes in a single package; thus, water inleakage is not considered. Reflection conditions of at least 20 cm of water at full density are assumed as this provides a conservative approximation of reflection conditions likely to be encountered. Since water inleakage is not assumed; only the package and contents need be considered in the development of the geometric condition of the package following the specified tests. Due credit may be taken in the specification of the geometric conditions in the criticality assessment for the condition of the package following the tests of paras 734(a) and 734(b) on separate specimens of the package. The conditions should be conservative but consistent with the results of the tests. Where the condition of the package following the tests cannot be demonstrated, worst case assumptions regarding the geometric arrangement of the package and contents should be made taking into account all moderating and structural components of the packaging. The assumptions should be in conformity with the potential worse-case effects of the mechanical and thermal tests, and all package orientations should be considered for the analysis. Subcriticality must be demonstrated after due consideration of such aspects as efficiency of moderator, loss of neutron absorbers, rearrangement of packaging components and contents, geometric changes, and temperature effects.

680.3. Subpara. (b) requires that, for the individual package, water leakage into or out of the package must be addressed unless the multiple water barriers are demonstrated to be watertight following the tests of paras 734 and 733. Thus, for packages transported by air the tests of 682(b) must be replaced with the tests of para. 680(b) in determining watertightness as required by para. 677(a).

680.4. In summary, subpara. (a) provides an additional assessment for a package transported by air while subpara. (b) provides a supplement to para. 677(a) to be applied in the assessment of para. 679 for packages transported by air.

Assessment of package arrays under normal conditions of transport

681.1. The assessment requires that all arrangements of packages be considered in the determination of the number of 5N packages that is subcritical because the neutron interaction occurring among the packages of the array may not be equal along the three dimensions.

681.2. The assessment might involve the calculation of large finite arrays for which there is a lack of experimental data. Therefore it is recommended that a specific supplementary allowance be made in addition to other margins usually allowed for random and systematic effects on calculated values of the neutron multiplication factor.

681.3. Note that "subcritical" means that the maximum neutron multiplication, adjusted appropriately by including a calculational bias, uncertainties, and a subcritical margin, should be less than 1.0. See Appendix VII for specific advice on the assessment procedure and advice on determining an upper subcritical limit..

Assessment of package arrays under accident conditions of transport

682.1. With the 1996 Edition of the Regulations, tests for the accident conditions of transport must consider the crush test of para. 727(c) for light-weight (<500 kg) and low-density (<1000 kg/m^3) packages. The criteria for invoking the crush test as opposed to the drop test of para. 727(a) is the same as that used for packages with contents greater than 1000A_2 not as special form (see para. 656(b)).

682.2. Subpara. (c) provides a severe restriction on any fissile material permitted to escape the package under accident conditions. All precautions to preclude the release of fissile material from the containment system should be taken. The variety of configurations possible for fissile material escaping from the containment system and the possibility of subsequent chemical or physical changes requires that the total quantity of fissile material that escapes from the array of packages should be less than the minimum critical mass for the fissile material type and with optimum moderator conditions and reflection by 20-cm of full-density water. An equal amount of material should be assumed to escape from each package in the array. The difficulty is in demonstrating the maximum quantity that could escape the containment system. Depending on the packaging components that define the containment and confinement systems, it is possible for fissile material to escape the containment system, but not the confinement system. In such cases there may be adequate mechanisms for criticality control. The intent of this subpara, however, is to ensure proper consideration of any potential escape of fissile material from the package where loss of criticality control must be assumed.

682.3. The assessment conditions considered should also include those arising from events less severe than the test conditions. For example, it is possible for a package to be subcritical following a 9m drop but to be critical under conditions consistent with a less severe impact.

682.4. See paras 681.1 through 681.3.

SECTION VII

TEST PROCEDURES

DEMONSTRATION OF COMPLIANCE

701.1. The Regulations contain performance standards, as opposed to specific design requirements. While this means greater flexibility for the designer it presents more difficulties in obtaining approval. The intent is to allow the applicant to use accepted engineering practice to evaluate a package or radioactive material. This could include the testing of full scale packages, scale models, mock-ups of specific parts of a package, calculations, and reasoned arguments or a combination of these methods. Regardless of the methods used, complete and proper documentation is of prime importance to satisfy the competent authority that all safety aspects and modes of failure have been considered. Any assumption should be clearly stated and fully justified.

701.2. Testing packages containing radioactive material presents a special challenge because of the radioactive hazard. While it may not be advisable to perform the tests required using radioactive material, it is necessary to convince the competent authority that the regulatory requirements have been met. When determining whether radioactive material or the intended radioactive contents are to be used in the tests, a radiological safety assessment should be made.

701.3. There are many other factors to consider in demonstrating compliance. These include but are not limited to the complexity of the package design, special phenomena that require investigation, the availability of facilities, and the ability to accurately measure and/or scale responses.

701.4. Where the Regulations require compliance with a specific leakage limit the designer should incorporate some means in the design to readily demonstrate the required degree of leak tightness. One method is to include some type of sampling chamber or test port that can be readily checked before shipment.

701.5. Test models should accurately represent the intended design with manufacturing methods and quality assurance and quality control similar to that intended for the finished product. Increased emphasis should be placed on the prototype to ensure that a test specimen is a true representation of production product. If simulated radioactive contents are being used these contents should truly represent the actual contents in mass, density, chemical composition, volume and any other characteristics which are significant. The contents should simulate any impact loads on the inside surface of the package and any closure lids. Any deficiencies or differences in the model should be documented prior to the testing and some evaluation done to determine how it may affect the outcome of the tests, either positively or negatively.

701.6. The number of specimens used in testing will be related to the design features to be tested and the desired reliability of the assessments. Repetition of tests with different specimens may be used to account for variations due to the range of properties in the material specifications or tolerances in the design.

701.7. The results of the tests may necessitate an increase in the number of specimens to meet the requirements of the test procedures in respect of maximum damage. It may be possible to use computer code simulations to reduce the number of tests required.

701.8. Care has to be exercised when planning the instrumentation and analysis of either a scale model test or a full scale test. It is necessary to ensure that adequate and correctly calibrated instrumentation and test devices are provided so that the test results may be documented and evaluated in order to verify the test results. At the same time, it is necessary to ensure that the instrumentation, test devices, and electrical connections do not interfere with the model in a way that would invalidate the test results.

701.9. When acceleration sensors are used to evaluate impact behaviour of the package, the cut-off frequency should be considered. The cut-off frequency should be selected to suit the structure (shape and dimension) of the package. Experience suggests that, for a package with a mass of 100 Mg with impact limiter, the cut-off frequency should be 100 to 200 Hz, and that, for smaller packages with a mass of m Mg, this cut-off frequency should be multiplied by a factor $(100/m)^{1/3}$. When the package includes components necessary to guarantee the safety under impact, and these components have a fundamental resonance or first mode frequencies exceeding the above mentioned cut-off value, the cut-off frequency may need to be adjusted so that the eliminated part of the signal has no significant influence on the assessment of the mechanical behaviour of these components. In these cases, a modal analysis may be necessary. Examples of such components include shells under evaluation for brittle fracture and internal arrangement structures needed for guaranteeing subcriticality. When such an issue is dealt with in an analytical evaluation, the calculation method and modelling should allow a pertinent assessment of these dynamic effects. This may require adjustment of the time steps and mesh size to low values consistent with the above mentioned frequencies used in the calculation.

701.10. In many cases it may be simpler and less expensive to test a full scale model rather than use a scale model or demonstrate compliance by calculation and reasoned argument. One disadvantage in relying completely on testing is that any future changes to either the contents or the package design may be much harder or impossible to justify. On a practical basis unless the packages are very inexpensive to construct and several are tested, it usually requires additional work to justify the test attitude.

701.11. When considering reference to previously satisfactory demonstrations of a similar nature, it is necessary to consider all the similarities and the differences between two packages. The areas of difference may require modification of the results of the demonstration. The ways and extent to which the differences and similarities will qualify the results from the previous demonstration depend upon their effects. In an extreme case, a packaging may be geometrically identical to that used in an approved package but because of material changes in the new packaging, the reference to the previous demonstration would not be relevant and should not be used.

701.12. Another method of demonstrating compliance is by calculation, or reasoned argument, when the calculation procedures and parameters are generally agreed to be reliable or conservative. Regardless of the qualification method chosen there will probably be a need to do some calculations and reasoned argument. Material properties in specifications are usually supplied to give a probability of not being under strength of between 95% to 98%. When tests are used for determining material property data, scatter in the data should be taken into account. It is usual to factor results where the number of tests are limited to give a limit of the mean plus twice the standard deviation on a normal (Gaussian) distribution (approximately 95% probability). It is also necessary to consider scatter due to material and manufacturing tolerances unless all calculations are on the worst combination of possible dimensions. When computer codes are used it should be made abundantly clear that the formulations used are applicable to finite deformation (i. e. not only large displacement but also large strain). In most cases the requirements, especially those involving accidental impact, will necessitate a finite strain formulation due to the potential severe damage inflicted. Ignoring such details could lead to significant error. In the presentation of reasoned argument, care is required that the argument is based on engineering experience.

Where theory is used, due account should be made of design details which could modify the result of general theory, e.g., discontinuities, asymmetries, irregular geometry, inhomogeneities and variable material properties. The presentation of reasoned argument is a difficult method because of its subjective nature.

701.13. Many calculations could require the use of commercially available computer codes. The reliability and the appropriate validation of the computer code selected must be considered. First, is the code applicable for the intended calculation? For example, for mechanical assessments, can it accept impact calculations? Is it suitable for calculating plastic as well as elastic deformations? Second, does the computer code adequately represent the packaging under review for the purpose of compliance? To satisfy these two requirements it may be necessary for the user to run 'bench mark' problems, which use the code to model and calculate the parameters of a problem in which the results are known. Options settings may have a strong influence on the validity of the benchmark studies to the problem being solved. In mechanical codes, options and modelling considerations include package material properties under dynamic conditions, elastic and plastic deformations, detailing connections between components such as screws and welds, and allowing friction, hydrodynamic, sliding and damping effects. User experience in the proper selection of code options, material properties and mesh selection can affect results using a particular code. Benchmark studies should also consider sensitivity of the results to parameter variation. Confidence can be increased by systematic benchmarking, proceeding from the simple to the complex. For other uses, checks that the input and output balance in load or energy may be required. When the code used is not widely employed and known, proof of the theoretical correctness should also be given.

701.14. Justification of the design may be done by the performance of tests with models of appropriate scale incorporating those features which are significant with respect to the item under investigation when engineering experience has shown results of such tests to be suitable for design purposes. When a scale model is used, the need for adjusting certain test parameters, such as penetrator diameter or compressive load, should be taken into account. On the other hand certain test parameters cannot be adjusted. For example, both time and gravitational acceleration are real, and therefore it will be necessary to adjust the results by use of scaling factors. Scale modelling should be supported by calculation or by computer simulation using benchmarked computer software to ensure that an adequate margin of safety exists.

701.15. When scale models are used to determine damage, due consideration should be given to the mechanisms which affect energy absorption since friction, rupture, crushing, elasticity, plasticity and instability may have different scale factors as a result of different parameters in the test being effected. Also, because the demonstration of compliance requires the combination of three tests (such as penetration, drop and thermal tests for Type B(U) and Type B(M) packages), conflicting requirements for the test parameters may require a compromise, which in turn would give results requiring scale factoring. In summary, the effect of scaling for all areas of difference should be considered.

701.16. Experience has shown that the testing of scale models may be very useful for demonstrating compliance with certain specific requirements of the Regulations, particularly the mechanical tests. Attempts to perform thermal tests using scale models are problematic (see para. 728.23 and 728.24). In mechanical tests, the conditions of similitude are relatively simple to create, provided the same materials and suitable methods of construction are used for the model as for the full sized package. Thus, in an economical manner, it is possible to study the relation of package orientation and the resulting damage, and the overall deformation of the package, and to obtain information concerning the deceleration of package parts. In addition, many design features can be optimized by model testing.

701.17. The details which should be included in the model are a matter of judgement and depend on the type of test for which the model is intended. For example, in the determination of the structural response from an end impact, the omission of lateral cooling fins from the scale model may result in more severe damage. This type of consideration may greatly simplify construction of the model without detracting from its validity. Only pertinent structural features which may influence the outcome of the test need be included. It is essential, however, that the materials of construction for the scale model and the full sized package are the same and that suitable construction and manufacturing techniques are used. In this sense, the construction and manufacturing techniques which will replicate the mechanical behaviour and structural response of the full sized package should be used, giving consideration to such processes as machining, welding, heat treatment and bonding methods. The stress-strain characteristics of the construction materials should not be strain rate dependent to a point which would invalidate the model results. This point needs to be made in view of the fact that strain rates in the model may be higher than in the full sized package.

701.18. In some cases it may not be practical to scale all components of the package precisely. For example, consider the thickness of an impact limiter compared to the overall length of the package. In the model, the ratio of the thickness to the overall length may differ from that of the actual package. Other examples include sheet metal gauge, gasket or bolt size that may not be standard size or may not be readily available. When any appreciable geometrical discrepancy exists between the actual package and the model to be tested, the behaviour of both when subjected to the 9 m drop should be compared by computer code analyses to determine whether the effect of geometrical discrepancy is a significant consideration. The computer code employed should be a code which has been verified through appropriate bench mark tests. If the effects of the discrepancies are not significant, the model could be considered suitable for a scale model drop test. This applies to a scale ratio of 1:4 or greater.

701.19. The scale factor chosen for the model is another area where a judgement needs to be made since the choice of scale factor depends on the accuracy necessary to ensure an acceptable model representation. The greater the deviation from full scale, the greater the error that is introduced. Consequently, the reduction of scale might be greater for a study of package deformation as a whole than for testing certain parts of the package and in some cases the scale factor chosen may be determined by the particular type of test being undertaken. In some tests, such as the penetration tests specified in the Regulations, it will also be necessary to scale the bar in order to produce accurate results. In other cases where the packaging may be protected by a significant thickness of deformable structure, the drop height may need to be scaled.

701.20. In general, the scale ratio M (the ratio model dimension:prototype dimension) should be not less than 1:4. For a model with a scale ratio of 1:4 or larger, the effect of strain rate dependence on the material mechanical properties will be negligibly small. The effect of strain rate dependence for typical materials (e.g., stainless steel) should be checked.

701.21. Scaling of drop tests is possible, taking into account the limitations given below, as a result of the following model laws, which are valid when the original drop height is maintained:

Accelerations: $a_{model} = (a_{original})/M$

Forces: $F_{model} = (F_{original})M^2$

Stresses: $s_{model} = s_{original}$

Strains: $0_{model} = 0_{original}$

701.22. For lightweight models, the model attitude or velocity during drop testing could be affected by such things as the swing of an umbilical cord carrying wires for acceleration sensors or strain gauges, or by wind effects. Experience suggests that, for packages with mass up to 1000 kg, full scale models should be used for the test, or special guides should be used with the scale model.

701.23. When an application for approval of a package design is based to any extent on scale model testing, the application should include a demonstration of the validity of the scaling methods used. In particular, such a demonstration should include:

- definition of the scale factor;
- demonstration that the model constructed reproduces sufficiently accurately the details of the package or packaging parts to be tested;
- a list of parts or features not reproduced in the model;
- justification for deletion of parts or features in the model; and
- justification of the similitude criteria used.

701.24. In the evaluation of the results of a scale model test, it is necessary to consider not only the damage sustained by the packaging, but, in some cases, the damage to the package contents. In particular, damage to the package contents should be considered when it involves a change in:

- release rate potential;
- parameters affecting criticality;
- shielding effectiveness;
- thermal behaviour.

701.25. It might be difficult to extrapolate the results of scale model testing involving seals and sealing surfaces to the responses expected in a full sized package. Although it is possible to acquire valuable information on the deformation and displacement of sealing surfaces with scale models, extrapolation of seal performance and leakage should be approached with caution (see para. 716.7). When scale models are used for testing seals it is necessary to consider the possible effect of such factors as surface roughness, seal behaviour as a function of material thickness and type, and the problems associated with predicting leakage rates on the basis of scale model results.

702.1. Any post-test assessment method used to assure compliance should incorporate the following techniques as appropriate to the type of package under examination:

- visual examination;
- assessment of distortion;
- seal gap measurements of all closures;
- seal leakage testing;
- destructive and non-destructive testing and measurement; and
- microscopic examination of damaged material.

702.2. In the evaluation of damage to a package after a drop test, all damage from secondary impacts should be considered as well. Secondary impact includes all additional impacts between the package and target, following initial impact. For evaluations which are based on numerical methods, it is also necessary to consider secondary impacts. Accordingly, the attitude of the package which produces maximum damage has to be determined with secondary as well as initial impacts taken into account. Experience suggests that the effect of secondary impact is often more severe for slender and rigid packages, including:

- a package with an aspect ratio (length to diameter) larger than 5, but sometimes even as low as 2;
- large package when significant rebound is expected to occur following the 9 m drop; and
- a package in which the contents are rigid and slender and particularly vulnerable to lateral impacts.

TESTS FOR SPECIAL FORM RADIOACTIVE MATERIAL

General

704.1. The four test methods specified in the Regulations, namely the impact, percussion, bending and heat tests, are intended to simulate mechanical and thermal effects to which a special form radioactive material might be exposed if released from its packaging.

704.2. These test requirements are provided to ensure that special form radioactive materials which become immersed in liquids as a result of an accident will not disperse more than the limits given in para. 603.

704.3. The tests of a capsule design may be performed with simulated radioactive material. The term `simulated' means a facsimile of a radioactive sealed source, the capsule of which has the same construction and is made with exactly the same materials as those of the sealed source that it represents but contains, in place of the radioactive material, a substance with mechanical, physical and chemical properties as close as possible to those of the radioactive material and containing radioactive material of tracer quantities only. The tracer should be in a form soluble in a solvent which does not attack the capsule. One procedure described in ISO 2919 [99] utilizes either 2 MBq of strontium-90 and yttrium-90 as soluble salt, or 1 MBq of cobalt-60 as soluble salt. When possible, shorter lived nuclides should be used. However, if leaching assessment techniques are used, care needs to be taken when interpreting the results. The effects of scaling will have to be introduced, the importance of which will depend upon the maximum activity to be contained within the capsule, its physical form, and in particular its solubility compared with the tracer radionuclide. These problems can be avoided if volumetric leakage tests are used (see paras. 603.3 and 603.4). Typically, tests for special form radioactive material are performed on full scale sealed sources or indispersible solid material because they are not expensive and the results of the tests are easy to interpret.

Test methods

705.1. Since this test is intended to be analogous to the Type B(U) package nine metre drop test (603.1), the specimen should be dropped so as to suffer maximum damage.

706.1. Special attention should be paid to the percussion test conditions in order to get maximum damage.

709.1. It is recognized that the tests indicated in paras 705, 706 and 708 are not unique and that other internationally accepted test standards may be equally acceptable. Two tests prescribed by the International Organization for Standardization have been identified as adequate alternatives.

709.2. The alternative test proposed in para. 709(a) is the ISO 2919 [100] Impact Class 4 test which consists of the following: A hammer, with a mass of 2 kg, the flat striking surface having a diameter of 25 mm, with its edge rounded to a radius of 3 mm, is allowed to drop on the specimen from a height of 1 m; the specimen is placed on a steel anvil which has a mass of at least 20 kg. The anvil is required to be rigidly mounted and has a flat surface large enough to take the whole of the specimen. This test may be employed in place of both the impact test (para. 705) and the percussion test (para. 706).

709.3. The alternative test proposed in para. 709(b) is the ISO 2919 [100] Temperature Class 6 test which consists of subjecting the specimen to a minimum temperature of -40EC for 20 minutes and heating over a period not exceeding 70 minutes from ambient to 800EC; the specimen is then held at 800EC for 1 hour followed by thermal shock treatment in water at 20EC.

Leaching and volumetric leakage assessment methods

711.1. For specimens which comprise or simulate radioactive material enclosed in a sealed capsule, either a leaching assessment or a volumetric leakage assessment method should be performed as follows:

(a) the leaching assessment is similar to those applied to indispersible solid material except the specimen is not immersed for seven days in water , but other steps remain the same, or,
(b) the alternative volumetric leakage assessment comprises any of the tests prescribed in ISO 9978 [57] which are acceptable to the competent authority

The latter assessment method allows a reduction of the test period due to the specimen not being immersed in water for seven days. Additionally some tests prescribed in ISO 9978 [57] are non-radioactive, therefore it implies a reduction in the period of time for using a shielded cell during the test, and consequently a considerable cost reduction.

TESTS FOR LOW DISPERSIBLE RADIOACTIVE MATERIAL

712.1. To receive relief from the Type C package requirements, low dispersible radioactive material (LDM) must meet the same performance criteria for impact and fire resistance as a Type C package without producing significant quantities of dispersible material.

712.2. To qualify as LDM, certain material properties have to be demonstrated by appropriate direct physical tests, by analytical methods or a proper combination of these. It has to be shown that, if the contents of a Type B(U) package or Type B(M) package were to be subjected to the required tests, it would meet the performance criteria laid down in para. 605. Three tests are required: the 90 m/s impact test onto an unyielding target, the enhanced thermal test and the leaching test. The impact and thermal test are non-sequential. For the leaching test the material has to be in such a form that it is representative of the material properties following either the mechanical or the thermal test. The tests to demonstrate the required LDM properties do not have to be performed with the entire package contents if the results obtained with a representative fraction of the package contents can be scaled up to the full package contents in a reliable way. That is, for instance, if the package contents consist of several identical items, and it can be shown that multiplying the release established for one such item by the total number of such items in a package gives an upper estimate for the whole package contents. For large items it is also possible to perform tests with an essential part of it, or with a scaled down model, as long as it is established how the test results obtained in this way can be extrapolated to the release behaviour of the entire package contents.

712.3. For the 90 m/s impact test it has to be demonstrated that the impact of the entire package contents, unprotected by the packaging, onto an unyielding target with a speed of at least 90 m/s would lead to an airborne release of radioactive material in gaseous or particulate form up to 100µm aerodynamic equivalent diameter (AED) of less than 100 A_2. The aerodynamic equivalent diameter of an aerosol particle is defined as the diameter of a sphere of density 1g/cm^3 which has the same sedimentation behaviour in air. The AED of aerosol particles can be determined by a variety of aerosol measuring instruments and techniques such as impactors, optical particle counters, and centrifugal separators (cyclones). Various experimental test procedures may be used. One possible approach is to impact a horizontally flying test specimen onto a vertical wall that has the required unyielding target attributes. All particulate matter with AED below 100µm that becomes airborne can be transported upward by an upward directed airstream of appropriate speed and then analysed according to particle size by established aerosol measuring techniques. An airstream with an upward speed of about 30cm/s would serve as a separator, in that particles with AED < 100µm would remain airborne, whereas larger particles would be removed due to their settling velocity exceeding 30 cm/s.

712.4. See paras. 605.5, 605.7-605.9 and 704.3 for additional information.

TESTS FOR PACKAGES

Preparation of a specimen for testing

713.1. Unless the actual condition of the specimen is recorded in advance of the test, it would be difficult to subsequently decide whether any defect has been caused by the tests.

714.1. Since, in certain cases, components forming a containment system may be assembled in different ways, it is essential for test purposes that the specimen and the method of assembly be clearly defined.

Testing the integrity of the containment system and shielding and assessing criticality safety

716.1. In order to establish the performance of specimens which have been subjected to the tests specified in paras 719-733 it may be necessary to undertake an investigation programme involving both inspection and further subsidiary testing. Generally, the first step will be a visual examination of the specimen and recording by photography. In addition, other inspections may be necessary. If the tests were performed with specimens containing radioactive trace material, wipe tests may give a measurement of the leakage. Leaktightness may be detected following the procedures outlined in paras 646.3 to 646.5 (Type IP, Type A, Type B). Likewise, the shielding integrity may be evaluated by the use of trace radiation materials placed inside the packaging. After examination of the outer integrity, the containment system should be disassembled to check the interior situation: integrity of capsules, glass, flasks, etc.; stability of geometrical compartments, particularly in the case where the intended contents are fissile material; distribution of absorbent material; stability of shielding; and function of mechanical parts. The investigatory programme should be aimed at examining three specific areas:

- integrity of the containment system;
- integrity of shielding;
- assurance, where applicable, that no rearrangement of the fissile contents or neutron poison or degree of moderation has adversely influenced the assumptions and predictions of the criticality assessment.

716.2. The integrity of the containment system can be evaluated in many ways. For example, the activity release from the containment system can be calculated on the basis of the volumetric (e.g., gaseous) release.

716.3. In the case of test specimens which are representative of full size containment systems, direct leakage measurements can be made on the test specimen.

716.4. The two following areas need attention:

- the performance of the normal closure system; and
- the leakage levels which may have occurred elsewhere in the containment system.

716.5. Containment, in accordance with the Regulations, involves so many variables that a single standard test procedure is not feasible.

In the American National Standard N14.5-1977 [31] types of tests which are acceptable, listed in order of increasing sensitivity under usual conditions, include but are not limited to:

- gas pressure drop
- water immersion bubble or soap bubble
- ethylene glycol
- gas pressure rise
- vacuum air bubble
- halogen detector
- helium mass spectrometer.

This standard

- relates the regulatory requirements for radioactive material containment to practical detectable mass flow leakage rates;
- defines the term `leaktight' in terms of a volumetric flow rate;
- makes some simplifying, conservative assumptions so that many of the variables may be consolidated;
- describes a release test procedure; and
- describes specific volumetric leakage tests.

ISO 12807 [32] specifies gas leakage test criteria and tests methods for demonstrating that Type B(U) and B(M) packages comply with the integrity containment requirements of the Regulations for design, fabrication, pre-shipment and periodic verifications. Preferred leakage test methods described by ISO 12807 include but are not limited to:

- Quantitative methods:
    - gas pressure drop
    - gas pressure rise
    - gas filled envelope-gas detector
    - evacuated envelope-gas detector
    - evacuated envelope-with back pressurization
- Qualitative methods:
    - gas bubble techniques
    - soap bubble
    - tracer gas-sniffer technique
    - tracer gas-spray method

This standard is mainly based on the following assumptions:

- Radioactive material could be released from the package in liquid, gas, solid, liquid with solids in suspension or particulate solid in a gas (aerosol), or any combination of such forms.
- Activity release or leakage can occur by one or more of the following ways: viscous flow, molecular flow, permeation or blockage.
- Radioactive contents release rate is measured indirectly by an equivalent gas leakage test in which it is measured gas flow rates (non-radioactive gas).
- Rates can be related mathematically to the diameter of a single straight capillary which in most cases is considered to conservatively represent a leak or leaks.

The main steps considered by the standard for determining leakage in both normal and accident conditions of transport are the following:

- Determination of permissible activity release rates
- Determination of standardized leakage rates
- Determination of permissible test leakage rates for each verification stage
- Selection of appropriate test methods; and
- Performance of tests and records of results.

716.6. If specimens less than full size have been used for test purposes, direct measurement of leakage past seals may not be advisable as not all parameters associated with leakage past seals are readily scaled. In this instance, because loss of sealing is often associated with loss of seal compression resulting from, for example, permanent extension of the closure cover bolts, it is recommended that a detailed metrology survey be made to establish the extent to which bolt extension and distortion of the sealing faces has occurred on the test specimen following the mechanical tests. The data based on a detailed metrology survey may be scaled and the equivalent distortion and bolt extension at full size determined. From tests with full size seals using the scaled metrology data the performance of the full size package may be determined.

716.7. For evaluating shielding integrity, ISO 2855 [77] draws attention to the fact that if a radioactive source is to be used to establish the post accident test condition, any damage or modification to the post-test package configuration caused by the insertion of the source might invalidate the results obtained.

716.8. If a full size specimen has been used for testing, one method of proving the integrity of the shielding is that, with a suitable source inside the specimen, the entire surface of the specimen be examined with an X-ray film or an appropriate instrument to determine whether there has been a loss of shielding. If there is evidence of loss of shielding at any point on the surface of the specimen, the radiation level should be determined by actual measurement and calculation to ensure compliance with paras 646, 651, 656 and 669. For additional information, refer to paras 646.1 to 646.5 and 656.13 to 656.18.

716.9. Alternatively, a careful dimensional survey could be made of those parameters which contribute to shielding performance to ascertain that they have not been adversely affected, e.g., slumping or loss of lead from shields, giving rise to either a general increase in radiation or increased localized radiation levels.

716.10. The applicable tests may demonstrate that the assumptions used in the criticality safety assessment are not valid. A change in the geometry or the physical or chemical form of the packaging components or contents could affect the neutron interaction within or between packages and any change should be consistent with the assumptions made in the criticality safety assessment of paras 671-682. If the conditions after the tests are not consistent with the criticality safety assessment assumptions, the assessment may need to be modified.

716.11. Although the testing of the package at full or smaller scale can be carried out with simulated contents from which some data on the behaviour of any basket or skip used for positioning the contents can be obtained, the final geometry will in practice depend upon the interaction of the actual material (whose mechanical properties may be different from the simulated contents) with both the basket or skip and the other components of the packaging.

Target for drop tests

717.1. The target for drop tests is specified as an essentially unyielding surface. This unyielding surface is intended to cause damage to the package which would be equivalent to or greater than that anticipated for impacts onto actual surfaces or structures which might occur during transport. The specified target also provides a method for assuring that analyses and tests can be compared and accurately repeated if necessary. The unyielding target, even though described in general terms, can be repeatedly constructed to provide a relatively large mass and stiffness with respect to the package being tested. So-called real targets, such as soil, soft rock and some concrete structures, are less stiff and could cause less damage to a package for a given impact velocity. In addition, it is more difficult to construct yielding surfaces that give reproducible test results, and the shape of the object being dropped can affect the yielding character of the surface. Thus, if yielding targets were used, the uncertainty of the test results would increase and the comparison between calculations and tests would be much more difficult.

717.2. One example of an unyielding target to meet the regulatory requirements is a 4 cm thick steel plate floated on to a concrete block mounted on firm soil or bedrock. The combined mass of the steel and concrete should be at least 10 times that of the specimen for the tests in paragraphs 705, 722, 725(a), 727 and 735, and 100 times that of the specimen for the test in paragraph 737, unless a different value can be justified. The steel plate should have protruding fixed steel structures on its lower surface to ensure tight contact with the concrete. The hardness of the steel should be considered when testing packages with hard surfaces. To minimize flexure the concrete should be sufficiently thick, but still allowing for the size of the test sample. Other targets which have been used are described in the literature [101],[102]. Since flexure of the target is to be avoided, especially in the vertical direction, it is recommended that the target should be close to cubic in form with the depth of the target comparable to the width and length.

Test for packagings designed to contain uranium hexafluoride

718.1. For the hydraulic test, only the cylinder is tested; valves and other service equipment should not be included in this leakage test. The valves and other service equipment should be tested consistent with ISO 7195 [26].

Tests for demonstrating ability to withstand normal conditions of transport

719.1. The climatic conditions to which a package may be subjected in the normal transport environment include changes in humidity, ambient temperature and pressure, and exposure to solar heating and rain.

719.2. Low relative humidity, particularly if associated with high temperature, causes the structural materials of the packaging such as timber to dry out, shrink, split and become brittle; direct exposure of a package to the sun can result in a surface temperature considerably above ambient temperature for a few hours around midday. Extreme cold hardens or embrittles certain materials, especially those used for joining or cushioning. Temperature and pressure changes can cause `breathing' and a gradual increase of humidity inside the outer parts of the packaging, and, if the temperature falls low enough, it can lead to condensation of water inside the packaging; the humidity in a ship's hold is often high and a fall in temperature will lead to considerable condensation on the outer surfaces of the package. If condensation occurs, fibreboard outer cases and spacers provided to reduce external radiation levels may collapse. Exposure to rain may occur while a package is awaiting loading or while it is being moved and loaded onto a conveyance.

719.3. A package may also be subjected to both dynamic and static mechanical effects during normal transport. The former may comprise limited shock, repeated bumping and/or vibration; the latter may comprise compression and tension.

719.4. A package may suffer a limited shock from a free drop onto a surface during handling. Rough handling, particularly rolling of cylindrical packages and tumbling of rectangular packages, is another common source of limited shock. It may also occur as a result of penetration by an object of relatively small cross-sectional area or by a blow from a corner or edge of another package.

719.5. Land transport often causes repeated bumping; all forms of transport produce vibrational forces which can cause metal fatigue and/or loose nuts and bolts. Stacking of packages for transport and any load movement as a result of a rapid change in speed during transport can subject packages to considerable compression. Lifting and a decrease in ambient pressure due to changes in altitude expose packages to tension.

719.6. The tests that have been selected to reproduce the kind of damage that could result from exposure to these climatic and handling/transport conditions and stresses are: the water spray test, the free drop test, the stacking test and the penetration test. It is unlikely that any one package would encounter all of the rough handling or minor mishaps represented by the four test requirements. The unintentional release of part of the contents, though very undesirable, should not be a major mishap because of the limitation on the contents of a Type A package. It is sufficient for each of three specimens to be subjected separately to the free drop, stacking and penetration tests, preceded in each case by the water spray test. However, this does not prevent one specimen from being used for all the tests.

719.7. The tests do not include all the events of the transport environment to which a Type A package may be subjected. They are, however, deemed adequate when considered in relation with the other general design requirements related to the transport environment, such as ambient temperature and its variation, handling and vibration.

720.1. If the water spray is applied from four directions simultaneously, it is necessary to allow for the time that the water, after the conclusion of this test, takes to gradually soak from the exterior into the interior of the package and lower its structural strength. Two hours was considered to be the appropriate interval on this basis. If the package is submitted to the succeeding free drop, stacking and penetration tests shortly after this interval, it will suffer the maximum damage. However, if the water spray is applied from each of the four directions consecutively, soaking of water into the interior of the package from each direction and drying of water from the exterior of the package will proceed progressively over a two hour period. Accordingly, no interval between the conclusion of the water spray test and the following free drop test should be allowed.

721.1. The water spray test is primarily intended for those packagings that rely on materials that absorb water or which are softened by water or materials bonded by water soluble glue. Packaging whose outer layers consist entirely of metal, wood, ceramic or plastic, or any combination of these materials, may be shown to pass the test by reasoned argument providing they don't retain the water and significantly increase their mass.

721.2. One method of performing the water spray test which is considered to satisfy the conditions prescribed in para. 721 is as follows:

(a) The specimen is placed on a flat horizontal surface, in whichever orientation is likely to cause most damage to the package. A uniformly distributed spray is directed onto the surface of the package for a period of 15 minutes from each of four directions at right angles and changes in spray direction should be made as rapidly as possible. More than one orientation may need to be tested.

(b) The following additional test conditions are recommended for consideration:
   - (i) A spray cone apex angle sufficient to envelop the entire specimen at the distance employed in (ii);
   - (ii) A distance from the nozzle to the nearest point on the specimen of at least 3 m;
   - (iii) A water consumption equivalent to the specified rainfall rate of 5 cm/h, as averaged over the area of the spray cone at the point of impingement on the specimen and normal to the centre line of the spray cone;
   - (iv) Water draining away as quickly as delivered;

(c) The requirement of para. 721 is intended to provide maximum surface wetting and this may be accomplished by directing the spray downwards at an angle of 45E from the horizontal:
   - (i) for rectangular specimens, the spray may be directed at each of the four corners;
   - (ii) for cylindrical specimens standing on one plane face, the spray may be applied from each of four directions at intervals of 90E.

It is recommended that the package not be supported above the surface to account for water that can be trapped at the base of the package.

722.1. The free drop test simulates the type of shock that a package would experience if it were to fall off the platform of a vehicle or if it were dropped during handling. In most cases packages would continue the journey after such shocks. Since heavier packages are less likely to be exposed to large drop heights during normal handling, the free drop distance for this test is graded according to package mass. If a heavy package experiences a significant drop it should be examined closely for damage or loss of contents or shielding. Light packages made from materials such as fibre board or wood require additional drops to simulate repeated impacts due to handling. It should be noted that, for packages containing fissile material, the requirement for additional free drop tests from a height of 0.3 m on each corner or, in the case of a cylindrical package, onto each quarter of each rim [para. 622(b) of the As Amended 1990 Edition of the Regulations] has been deleted from the 1996 Edition because such packages of metallic construction are not considered vulnerable to cumulative damage in the same way as certain lightweight wooden or fibreboard packages. Any inadequacies in a fissile package design with respect to its ability to withstand normal handling would be revealed by the test of para.722. The additional 0.3 m free drop tests still apply to certain wooden or fibreboard packages, in the 1996 Edition of the Regulations, whether or not they contain fissile material. This introduces a measure of consistency into the package testing regime.

722.2. Any drop test should be conducted with the contents of the package simulated to its maximum weight. More than one drop may be necessary to evaluate all possible drop attitudes. It may also be necessary to test specific features of the package such as hinges or locks to ensure containment, shielding and nuclear criticality safety are maintained.

722.3. The features to be tested depend on the type of the package to be tested. Such features include structural components, materials and devices designed to prevent loss or dispersal of radioactive substances or loss of shielding materials, (e.g., the entire containment system, such as lids, valves and their seals). For packages containing fissile materials, the features could include, in addition to those mentioned above, components for maintaining subcriticality, such as a fuel holding frame and neutron absorbers.

722.4. The `maximum damage' is the maximum impairment of the integrity of the package. To produce the `maximum damage' for most packages, the specimen should be dropped in one or more attitudes in such a way that the impact acceleration and/or deformation of the components under consideration is maximized. Most containers have some asymmetry giving different resistances to impact. In any investigation, sufficient structural elements should be considered to allow for the absorption of all the kinetic energy of the package. Arguments should be developed as to the damage in the various elements between the impact point and the concentration of mass with regard to their performance in absorbing the energy, in developing internal loads, in distorting, collapsing or folding, and in the consequences of these behaviours.

722.5. Packages of low mass could be hand held above the target and dropped, providing the desired attitude can be maintained. Mechanical means should be devised to hold and release the package in the desired impact attitude. This could be simply a release mechanism suspended from an overhead structure, like a roof member or a crane, or a tower specially designed for drop tests. The design of dedicated drop facilities has four main elements: the support, the release, the track guide (usually not used in direct drops), and the target which is defined in para. 717. Sufficient height is required in the support to allow for the release mechanism, the support cable or harness and the full depth of the test item and still make it possible to attain the correct attitude and dropping height between the bottom of the package and the target. In the case where a package has impact limiters the lowest point of the impact limiter would be used to determine the drop height. The release mechanism for a free drop test should, allow easy setting and instantaneous release, not give undesirable effects on the attitude of the specimen, and not add to the mechanical damage of the specimen. Various types of mechanisms, such as mechanical, electromagnetic, or combinations of these could be used. A number of test facilities are described in IAEA TECDOC-295 [103] and in the "Directory of test facilities for radioactive materials transport packages" published in the International Journal of Radioactive Materials Transport [104].

722.6. During the revision process leading to the 1996 edition of the IAEA regulations, it was agreed that all possible drop test orientations need not be considered when conducting the normal condition of transport drop test. Providing that it was not possible in "normal" conditions for the package to be dropped in certain orientations, these orientations could be ignored in assessing the worst damage. It was envisaged that this relaxation would only be allowed for large dimension and large aspect ratio packages. In addition this relief would require documented justification by the package designer. Package designs requiring competent authority approval should be tested in the most damaging drop test attitudes irrespective of package size or aspect ratio.

722.7. Scale model techniques may be useful in order to determine the most damaging drop attitude (see paras 701.7 to 701.25). Care should be taken in instrumentation since mounts and sensor frequencies may produce errors in the data obtained.

723.1. The stacking test is designed to simulate the effect of loads pressing on a package over a prolonged period of time to ensure that the effectiveness of the shielding and containment systems will not be impaired and, in the case of the contents being fissile material, will not adversely affect the configuration. This test duration corresponds to the requirements of the United Nations Recommendations [7].

723.2. Any package whose normal top, i.e., the side opposite the one which it normally rests on, is parallel and flat, could be stacked. In addition, stacking could be achieved by adding feet, extension pads or frames to the package with convex surfaces. Packages with convex surfaces cannot be stacked unless extension pads or feet are provided.

723.3. The specimen should be placed with the base down on an essentially flat surface such as a flat concrete floor or steel plate. If necessary, a flat plate, which has sufficient area to cover the upper surface of the specimen, should be placed on the upper surface of the specimen so that the load may be applied uniformly to it. The mass of the plate should be included in the total stacking mass being applied. If a number of packages of the same kind are stackable, a simple method is to build a stack of five packages on top of the test specimen. Alternatively, a steel plate or plates or other convenient materials with a mass five times that of the package may be placed on the package.

724.1. The penetration test is intended to ensure that the contents will not escape from the containment system or that the shielding or confinement system would not be damaged if a slender object such as a length of metal tubing or a handlebar of a falling bicycle should strike and penetrate the outer layers of the packaging.

Additional tests for Type A packages designed for liquids and gases

725.1. These additional tests for a Type A package designed to contain liquids or gases are imposed because liquid or gaseous radioactive material has a greater possibility of leakage than solid material. These tests do not require the water spray test first.

Tests for demonstrating ability to withstand accident conditions of transport

726.1. The accident tests specified in the Regulations were originally developed to satisfy two purposes. First, they were conceived as producing damage to the package equivalent to that which would be produced by a very severe accident (but not necessarily all conceivable accidents). Second, the tests were stated in terms which provided the engineering basis for the design. Since analysis is an acceptable method of qualifying designs, the tests were prescribed in engineering terms which could serve as unambiguous, quantifiable input to these calculations. Thus, in the development of the test requirements attention was given to how well these tests could be replicated (see, for example, para. 717.1).

726.2. The 1961 Edition of the Regulations was based on the principle of protection of the package contents, and hence the public health, from the consequences of a `maximum credible accident'. This phrase was later dropped because it did not give a unique level or standard with which to work and which was necessary to ensure the international acceptability of unilaterally approved designs. Recognition of the statistical nature of accidents is now implicit in the requirements. A major aim of the package tests is international acceptability, uniformity and repeatability; tests are designed so that the conditions can be readily reproduced in any country. The test conditions are intended to simulate very severe accidents in terms of the damaging effects on the package. They will produce damage exceeding that arising in the vast majority of incidents recorded, whether or not a package of radioactive material was involved.

726.3. The purpose of the mechanical tests (para. 727) and the thermal test (para. 728) that follow is to impose on the package damage equivalent to that which would be observed if the package were involved in severe accidents. The order and type of tests are considered to correspond to the order of environmental threat to the packaging in a real transport accident, i.e., mechanical impacts followed by thermal exposure. The test sequence also ensures mechanical damage to the package prior to the imposition of the thermal test; thus the package is most liable to sustain maximum thermal damage. The mechanical and thermal tests are applied to the same specimen sequentially. The immersion test (para. 729) may be conducted on a separate specimen because the probability of its occurrence in conjunction with a thermal/mechanical accident is extremely low.

727.1. Mechanical test requirements for Type B packages were introduced in the 1964 Edition of the Regulations, replacing the requirement of withstanding a `maximum credible accident', which was not specified by specific test requirements but left to the competent authority of the country concerned. Since Type B(U) and Type B(M) packages are transported by all modes of transport, the Type B(U) and Type B(M) test requirements are intended to take into account a large range of accidents which can expose packages to severe dynamic forces. The mechanical effects of accidents can be grouped into three categories: impact, crush and puncture loads. Though the figures for the test requirements were not derived directly from accident analyses at that time, subsequent risk and accident analyses have demonstrated that they represent very severe transport accidents [105],[106],[107],[108],[109],[110].

727.2. In drop I, the combination of the 9 m drop height, unyielding target and most damaging attitude produce a condition in which most of the drop energy is absorbed in the structure of the packaging. In real transport accidents, targets such as soil or vehicles yield, absorbing part of the impact energy and only higher velocity impacts may create equivalent damage [108],[109],[110].

727.3. Thin walled packaging designs or designs with sandwich walls could be sensitive to puncture loads with respect to loss of containment integrity, loss of thermal insulation or damage to the confinement system. Even thick walled designs may have weak points such as closures of drain holes, valves, etc. Puncture loads could be expected in accidents as impact surfaces are frequently not flat. In order to provide safety against these loads, the 1 m drop test onto a rigid bar was introduced. The drop height and punch geometry parameters are more the result of an engineering judgement than deductions from accident analyses.

727.4. The degree of safety provided by the 9 m drop test is smaller for light, low density packages than for heavy, high density packages, owing to the reduced impact energy and to the increased probability of impacting a relatively unyielding `target' [108],[111],[112],[113],[114] . Such packages may also be sensitive to crush loads. Accident analyses show that the probability of dynamic crush loads in land transport accidents is higher than that of impact loads because lightweight packages are transported in larger numbers or together with other packages [105],[106],[107]. Also, handling and stowage mishaps can lead to undue static or dynamic crush loads. The end result of this was the inclusion of the crush test (drop III) in the 1985 Edition of the Regulations. Packages containing a large amount of alpha emitters are generally light-low density packages because of the requirement for less shielding, and may fit into this category. This includes, for example, plutonium oxide powders and plutonium nitrate solutions, which are radioactive materials with high potential hazards. Because of their physical characteristics, most packages will be subject to the 9 m drop (impact) test rather than the crush test.

727.5. The Regulations require that the attitudes of the package for both the impact (drop I) or crush (drop III) and the penetration (drop II) tests be such as to produce maximum damage taking into account the thermal test. In addition, the order in which the tests are carried out is that which will be most damaging. The assessment of maximum damage should be made with concern for the containment of the radioactive material within the package, the retention of shielding to keep external radiation to the acceptable level and, in case of fissile materials, maintenance of subcriticality. Any damage which would give rise to increased radiation or loss of containment or affect the confinement system after the thermal test should be considered. Damage which may render the package inappropriate for re-use but does not affect its ability to meet the safety requirements should not be a reason for classifying the specimen as having failed.

727.6. Different modes of damage are possible as a result of the mechanical tests. It is necessary to consider the results of these modes for any analytical assessment to demonstrate compliance with the applicable requirements. The fracture of a critical component or the breach of the containment system may allow the escape of the radioactive material. Deformation may impair the function of radiation or thermal shields and it may alter the configuration of fissile material and should be reflected in the assumptions and predictions in the criticality assessment. Local damage to shielding may, as a result of the subsequent thermal test, give rise to deterioration of both the thermal and radiation protection. Consequently, investigations should include stress, strain, instability and local effect for all attitudes of drop where symmetry does not prevail.

727.7. Multiple drops of a specimen for the same test may not be feasible because of previous damage. It may be necessary to use more than one test sample or use analysis and reasoned argument based on engineering data to predict the most damaging attitude and to eliminate testing those attitudes where the safety is not impaired.

727.8. The most severe attitudes for symmetric packagings that have either a cylindric or cubic form may often be determined by the use of published information [103],[115]. Asymmetries, especially where protrusions occur, are often sensitive when used as the impact point. Lifting and handling devices such as skids or attachment points will often have a different strength or stiffness relative to the adjacent parts of the package and should be considered as possible impact points.

727.9. Discontinuities such as the lid or other penetration attachments could give a locally stiff element of structure of limited strength which could fail by either adjacent structural deformation or high loading (due to decelerations) on their retained masses.

727.10. Thin wall packages, such as drums, should be considered in terms of the possibility of plastic deformation either causing loss of the containment seal or distorting the lid attachment sufficiently to allow the loss of the lid.

727.11. Para. 671 requires that, for fissile materials, criticality analyses be made with the damage resulting from the mechanical and thermal test included. Consideration is required of such aspects as efficiency of moderator, loss of neutron absorbers, rearrangement of package contents, geometric changes and temperature effects. The assumptions made in the criticality analysis should be in conformity with the effects of the mechanical and thermal tests and all package orientations should be considered for the analysis.

727.12. It is intended that the drop of the package (drops I and II) or of the 500 kg mass (drop III) should be a free fall under gravity. If, however, some form of guiding is used, it is important that the impact velocity should be at least equal to the impact velocity where the package or the mass is under free fall (approx. 13.3 m/s for drops I and III).

727.13. For drop II, the required minimum length of the penetrating bar is 20 cm. A longer bar length should be used when the distance between the outer surface of a package and any important inner component for the safety of the package is greater than 20 cm or when the orientation of the model requires it. This is particular true for specimens with large impact limiting devices where the penetration can be considerable. The material specified for the construction of the bar is mild steel. The minimum yield stress of such material should not be less than 150 MPa nor more than 280 MPa. The yield to ultimate stress ratio should not be greater than 0.6. It may be difficult to perform a test where buckling of the bar is possible. In this case, justification of the bar length to obtain maximum damage to the specimen should be done.

727.14 For drop II, the most damaging package orientation is not necessarily a flat impact onto the bar top surface. For some package designs it has been shown that oblique orientations at angles in the range of 20° to 30° cause maximum damage due to the initiation of penetration of the bar corner into the external envelope of the package.

727.15. For preliminary design purposes only, for the outer shell of a steel-lead-steel packaging, the following equation may be used to estimate the shell thickness required to resist failure when the package is subjected to the penetration test:

$$t = 2148.5 \left(\frac{w}{s} \right)^{0.7}$$

where,

t is the outer shell thickness (cm)
w is the mass of the package (kg)
s is the tensile strength of the outer shell material (Pa).

This equation is based on tests employing annealed mild steel backed by chemical lead (see ORNL-NSIC-68 [115]). Packages using materials having different physical properties could require different thicknesses of the outer steel shell to meet the requirements. For packages having small diameters, less than 0.75 m, or using materials having different physical properties, or for impacts near changes of geometry or in oblique attitudes, the preliminary estimate may not be conservative [115].

727.16. For the crush test (drop III) the packaging should rest on the target in such a way that it is stable. In order to achieve this it may be necessary to provide support, in which case the presence of the support should not influence the damage to the package [116].

727.17. Instrumentation of test specimens and even of the target response to impact should be done for the following reasons:

- validation of assumptions in the safety analysis
- as a basis for design alterations
- as a basis for the design of comparable packages
- as a benchmark test for computer codes.

727.18. Examples of functions that should be measured under impact/crushing conditions are: deceleration-time function and strain-time function. Where electronic devices are used to acquire, record and store data, examination of any filtering, truncating or cropping should be made so that no data peaks of significance are lost. Most instruments will require cable connections to external devices. These connections should be such that they neither restrict the free fall of the package nor restrain the package in any way after impact (see para. 701.9).

728.1. Work in the United States of America [105],[106],[107],[117],[118],[119],[120] suggests that the thermal test specified in para. 728 provides an envelope of environments which encompasses most transport related accidents involving fires. The Regulations specify a test condition based on a liquid hydrocarbon-air fire with a duration of 30 minutes. Other parameters relating to fire geometry and heat transfer characteristics are specified in order to define the heat input to the package.

728.2. The thermal test specifies a liquid hydrocarbon pool fire which is intended to encompass the damaging effects of fires involving liquid, solid or gaseous combustible materials. Liquids such as liquid petroleum gas (LPG) or liquid natural gas (LNG), and liquid hydrogen are covered by the test because pool fires with such fuels generally will not last for 30 minutes. Liquid petroleum products are frequently transported by road, rail and sea and would be expected to give rise to a fire following an accident. Liquids that can flow around the package and create the stipulated conditions are restricted to a narrow range of calorific values, so the severe fire is quite well defined.

728.3. The flame temperature and emissivity (800EC and 0.9) define time and space averaged conditions found in pool fires. Locally, within fires, temperatures and heat fluxes can exceed this description. However, non-ideal positioning of a package within a fire, the movement with time of the fire source relative to the package, shielding by other non-combustible packages or conveyances involved in the accident, wind effects and the massive structure of many Type B(U) and Type B(M) packages will all combine to average the conditions to conform to, or be less severe than, the test description [119],[120]. The presence of a package and remoteness from the oxygen supply (air passing through about 1 m of flame) may both tend to depress the flame temperature adjacent to the package. Natural winds can supply extra oxygen but tend to remove flame cover from parts of the package, hence the requirement of quiescent ambient conditions. Use of a vertical flame guide underneath the package will minimize the effect of wind and improve flame coverage [121]. The flame emissivity is difficult to assess, as direct measurements are not generally available, but indications from practical tests suggest that the 0.9 value specified is an overestimate. The combination of parameters in the test results in a severe flame description unlikely to be exceeded by accident conditions.

728.4. The duration of a large petroleum fire depends on the quantity of fuel involved and the availability of fire fighting resources. Liquid fuel is carried in large quantities but, in order to form a pool, any leakage must flow into a well defined area around the package with consequent losses by drainage. In general, not all the contents of a single tank will be involved in this way as much will be consumed either in the tank itself or during transfer to the vicinity of the package. The contents of other tanks will most likely be burnt at a more remote location as the fire moves from tank to tank. Recognition must also be given to the fact that, when lives are not directly at risk, fires are often allowed to continue to natural extinction. Consequently, historical records of fire durations should be viewed critically. The 30 minute duration is therefore chosen from consideration of these factors and encompasses the low probability of a package being involved in a fire with a large volume of fuel and the `worst case' geometry specified. The low probability, long duration fire is most likely to occur in combination with a geometry which effectively reduces the thermal input, with the package resting on the ground and/or protected by the vehicle structure. The heat input from the thermal test is thus consistent with realistic, severe accident situations.

728.5. The following configuration for the fire geometry minimizes the effects of radiation losses and maximizes heat input to the packages. A 0.6 to 1 m elevation of the package ensures that the flames are well developed at the package location, with adequate space for the lateral in-flow of air. This improves flame uniformity without affecting the heat fluxes. The extension of the fuel source beyond the package boundary ensures a minimum flame thickness of about 1 m, providing a reasonably high flame emissivity. The size of the pool should be between 1 and 3 m beyond any external surface of the test specimen to improve flame coverage. Larger extensions can lead to oxygen starvation at the centre and relatively low temperatures close to the package [122].

728.6. Previous editions of the Regulations have required that no artificial cooling should be used before three hours have expired following cessation of the fire. The 1985 Edition deleted reference to the three hour period, implying that the assessment of temperatures and pressures should continue until all temperatures, internal and external, are falling and that natural combustion of package components will continue without interference. Only natural convection and radiation contribute to heat loss from the package surface after the end of the fire.

728.7. The Regulations allow other values of surface absorptivity to be used as an alternative to the standard value of 0.8 if they can be justified. In practice, a pool fire is so smoky that it is probable that soot will be deposited on cool surfaces, modifying conditions there. This is likely to increase the absorptivity but interpose a conduction barrier. The value of 0.8 is consistent with thermal absorptivities of paints and can be considered as approximating the effects of surface sooting. As a surface is heated, the soot may not be retained and lower values of surface absorptivity could result.

728.8. The 1985 Edition of the Regulations removed the previous ambiguity of ``convection heat input in still ambient air at 800EC'' but did not specify a value for the coefficient, requiring the designer to justify the assumptions. A significant proportion of the heat input may derive from convection, particularly when the outer surface is finned and early in the test when the surfaces are relatively cool. The convective heat input should be at least equivalent to that for a hydrocarbon fuel air fire at the specified conditions.

728.9. The effects of the thermal test are, of course, dominated by increased package temperatures and consequent effects such as high internal pressures. The peak temperature depends to some extent on the initial temperature, which should therefore be determined using the highest appropriate initial conditions of internal heat generation, solar heating and ambient temperature. For a practical test, not all of these initial conditions will be achievable, so appropriate measurements (e.g., ambient temperature) should be made, and package temperatures corrected after the test.

728.10. The fire conditions defined in the Regulations and the requirement for full engulfment for the duration of the test represent a very severe test of a package. It is not intended to define the worst conceivable fire. In practice, some parameters may be more onerous than specified in the Regulations but others would be less demanding. For example, it is difficult to conceive of a practical situation where all surfaces of a package could experience the full effects of the fire, since it would be expected that a significant fraction of the surface area would be shielded, either by the ground or wreckage and debris arising from the accident. Emphasis has been placed on the thermal heat flux rather than on the individual parameters chosen, and in this respect the conditions specified represent a very severe test for any package [120]. It should also be emphasized that the thermal test is only one of a cumulative series of tests which must be applied to yield the maximum damage in a package. This damage must remain demonstrably small in terms of stringent criteria of containment integrity, external radiation level and nuclear criticality safety.

728.11. The following are examples of what are generally acceptable to competent authorities; other methods or techniques may be used, but more justification might be expected in support of such an approach. It is important to note that the requirements of the thermal test may be met by a practical test, by a calculated assessment, or by a combination of both. The last approach may be necessary, if, for example, the initial conditions required for a practical test were not achieved or if all the package design features were not fully represented in the experiment. In many cases, the consequences of the thermal test need to be determined by calculation, which therefore becomes an integral part of the planning and execution of the practical test. The Regulations specify certain fire parameters which are essential input data for the calculation method but which are generally uncontrollable parameters in practical tests. Standardization of the practical test is therefore achieved by defining the fuel and test geometry for a pool fire and requiring other practical methods to provide the same or greater heat input.

728.12. With regard to the package design, some shielding materials have eutectics with melting temperatures which are lower than the 800EC environment of the thermal test. Therefore consideration should be given to the capability of any structural materials to retain them. Local shielding materials such as plastics, paraffin wax or water may vaporize, causing a pressure which may rupture a shell which may have been weakened by damage from the mechanical tests. A thermal analysis may be required to determine whether such pressures can be attained.

728.13. The bottom of the package to be tested should be between 0.6 and 1 m above the surface of the liquid fuel source. Unless the fuel is replenished, or replaced by another liquid such as water, the level will fall during the test, probably by about 100 to 200 mm. The specimen package should be supported in such a way that the flow of heat and flames is perturbed by the minimum practical amount. For example, a larger number of small pillars is to be preferred to a single support covering a large area of the package. The transport vehicle, and any other ancillary equipment which might protect the package in practice, should be omitted from this test as the protection was taken into account in the test definition.

728.14. The pool size should extend between 1 and 3 m beyond the edges of the package so that all sides of the package are exposed to a luminous flame not less than 0.7 m and not more than 3 m thick, taking into account the reduction of the flame thickness with increasing height over the pool. In general, larger packages will require a larger extension as flame thicknesses will vary more over the greater distances involved. The requirement for fully engulfing flames can be interpreted as a need for all parts of the package to remain invisible throughout the 30 minute test, or at least for a large proportion of the time. This is best achieved by designing for thick flame cover which can accommodate natural variations in thickness without becoming transparent. A low wind velocity (quiescent conditions) is also required for stable flame cover, although large fires might generate high local wind velocities. Wind screens or baffles can help to stabilize the flames, but care should be taken to avoid changing the character of the flames and to avoid reflected or direct radiation from external surfaces. This would enhance the heat input and therefore not invalidate the test, but could make it more stringent than necessary.

728.15. Wind speeds of less than about 2 m/s should not detract from the test and short duration gusts of higher speeds will not have a large effect on high heat capacity packages particularly if flame cover is maintained. Open air testing should only take place when rain, hail or snow will not occur before the end of the post-fire cool-down period. The package should be mounted with the shortest dimension vertical for the most uniform flame cover, unless a different orientation will lead to a higher heat input or greater damage, in which case such an arrangement should be chosen.

728.16. The fuel for a pool fire should comprise a distillate of petroleum with a distillation end point of 330EC maximum and an open cup flash point of 46EC minimum and with a gross heating value of between 46 and 49 MJ/kg. This covers most hydrocarbons derived from petroleum with a density of less than 820 kg/m^3, e.g., kerosene and JP4-type fuels. A small amount of more volatile fuel may be used to ignite the pool as this will have an insignificant effect on the total heat input.

728.17. The choice of instrumentation will be dictated by the use to be made of a practical thermal test. Where a test provides data to be used in calculations to demonstrate compliance, some instrumentation is essential. The type and positioning of the instruments will depend on the data needed, e.g., internal pressure and temperature measurements may be necessary and, where stress is considered important, strain gauges should be installed. In all cases, the cables carrying signals through the flames should be protected to avoid extraneous voltages created at high temperatures. As an alternative to continuous measurement, the package might be equipped in such a way that instruments could be connected soon after the fire and early enough to measure the peak pressure and temperature. A measurement of leakage can be achieved by pre-pressurization and re-measurement after the thermal test, where necessary making appropriate adjustments for temperature (see paras 656.5 to 656.24).

728.18. The duration of the test can be controlled by providing a measured supply of fuel calculated to ensure the required 30 minute duration, by removing the supply of fuel a predetermined time before the end of the test, by discharging the fuel from the pool at the end of the test or by carefully extinguishing the fire without affecting the package surfaces with the extinguishing agent. The duration of the test is the time between the achievement of good flame cover and required flame temperatures, and the time at which such cover and temperature is lost.

728.19. Measurements should continue after the fire, at least until the internal temperatures and pressures are falling. If rain, or other precipitation, occurs during this period a temporary cover should be erected to protect the package and to prevent inadvertent extinguishing of combustion of the package materials, but care should be taken not to restrict heat loss from the package.

728.20. Where the test supplies data for an analytical evaluation of the package, measurements made during the test should be corrected for non-standard initial conditions of ambient temperature, insolation, internal heat load, pressure, etc. The effects of partial loading, i.e., less than full contents, on the package heat capacity and heat transfer should be assessed.

728.21. A furnace test is often more convenient than an open pool fire test. Other possible test environments include pit fires and an open air burner system operating with liquified petroleum gas [123]. Any such test is acceptable provided it meets the requirements of para. 728. Methods to verify the required heat input and methods to prove the thermal environment can be found in the literature [124],[125],[126],[127].

728.22. Ensuring that the internal temperature increase is not less than that predicted for an 800EC fire ensures that the heat input is satisfactory, but the test should continue for at least 30 minutes, during which the time-averaged environment temperature should be at least 800EC. A high emissivity radiation source should be created by selecting a furnace either with an internal surface area very much larger than the envelope area of the package or with an inherently high emissivity internal surface (0.9 or higher). Many furnaces are unable to reproduce either the desired emissivity or the convective heat input of a pool fire, so an extension of the test duration might be necessary to compensate. Alternatively, a higher furnace temperature can be used but the test duration should be a minimum of 30 minutes. The furnace wall temperature should be measured at several places, sufficient to show that the average temperature is at least 800EC. The furnace can be pre-heated for a sufficient time to achieve thermal equilibrium, so avoiding a large temperature drop when the package is inserted. The 30 minute minimum duration should be such that the time-averaged environment temperature is at least 800EC.

728.23. The calculation of heat transfer or the determination of physical and chemical changes of a full size package based on the extrapolation of the results from a thermal test of a scale model may be impossible without many different tests. A wide ranging programme simulating each process separately would require an extensive investigation using a theoretical model, so the technique has little inherent advantage over the normal analytical approach. Any scale testing, and the interpretation of the results, should be shown to be technically valid. However, the use of full scale models of parts of the package might be useful if calculation for a component (such as a finned surface) proves difficult. For example, the efficiency of a heat shield, or of a shock absorber acting in this role, could be most readily demonstrated by a test of this component with a relatively simple body beneath it. Component modelling is of importance for the validation of computer models. However, measurements of flame temperature and flame and surface emissivities are difficult and might not provide a sufficiently accurate specification for a validation calculation. Component size should be selected and appropriate insulation provided so that heat entering from the artificial boundaries (i.e., those representing the rest of the package) is not significant.

728.24. Thermal testing of reduced scale models meeting the specified conditions of the thermal test may be performed and lead to conservative results of temperatures assuming that there is no fundamental change in the thermal behaviour of the components.

728.25. The most common method of package assessment for the thermal test is calculation. Many general purpose, heat transfer computer codes are available for such package modelling, although care should be taken to ensure that the provisions available in the code, in particular for representing radiation heat transfer from the environment to external surfaces, are adequate for the package geometry. Practical tests may ultimately be required for validation but arguments showing that the approximations or assumptions produce a more stringent test than required are often used. In general, code validation is accomplished by comparison with analytical solutions and comparison with other codes.

728.26. Generally, the normal conditions of transport will have been assessed by calculation, so detailed temperature and pressure distributions should be available. Alternatively, the package temperatures might have been measured experimentally, so, after correction to the appropriate ambient temperature, and for the effects of insolation and the heat load due to the contents, these provide the initial conditions for the calculated thermal test conditions. Ambient temperature corrections can be made in accordance with para. 651.4.

728.27. The external boundary conditions of the fire should represent radiation, reflection and convection. The temperature is specified by the Regulations as an average of 800EC so, in general, a uniform average temperature of 800EC should be used for the radiation source and for convective heat transfer.

728.28. The flame emissivity is prescribed as 0.9. This can be used without ambiguity for plane surfaces but, for finned surfaces, the thin flames between the fins will have an emissivity much lower than that value. The dominant source of radiation to the finned surfaces will therefore be the flames outside the fins; radiation from flames within the fin cavity can be ignored. In all cases, appropriate geometric view factors should be used with the fin envelope radiation source, and reflected radiation should be taken into account. Care should be taken to avoid the inclusion of radiation `reflected' from a surface representing flames as this is a non-typical situation.

728.29. The surface absorptivity is prescribed as 0.8 unless an alternative value can be established. In practice, demonstration of alternative values will be extremely difficult as surface conditions change in a fire, particularly as a result of sooting, and evidence obtained after a fire may not be relevant. The value of 0.8 is therefore most likely to be used in analytical assessments. It is important to take into account reflected radiation, particularly with complex finned surfaces, as multiple reflections increase the effective absorptivity to near unity. This complexity can be avoided by assuming unity for the surface absorptivity but, even in this case, surface to surface radiation should not be ignored, particularly during the cool-down period.

728.30. Convection coefficients during the fire should to be justified. Pool fire gas velocities are generally found to be in the range of 5 to 10 m/s [128]. Use of such velocities in forced convection, heat transfer correlations (e.g., the Colburn relation u = 0.036 $Pr^{1/3}$ $Re^{0.8}$ quoted by McAdams [129]) results in convective heat transfer coefficients of about 10 W/m^2C for large packages. Natural convection coefficients (about 5 W/m^2C) are not appropriate as this implies downward gas flow adjacent to the cool package walls, whereas, in practice, a general buoyant upward flow will dominate. The upper surface of a package is unlikely to experience such high gas velocities, in quiescent atmospheric conditions, as the region will include a stagnation area in the lee of the upward gas flow. The reduced convection there is adequately represented by the average coefficient as the averaging process includes this effect.

728.31. Convection coefficients for the post-test, cool-down period can be obtained from standard natural convection references, e.g., McAdams [129]. In this case coefficients appropriate for each surface can readily be applied. For vertical planes the turbulent natural convection equation is

$$Nu = 0.13\ (Pr\ Gr)^{1/3}$$

for Grashof numbers > 10^9. The boundary conditions used for the assessment of conditions under normal operation should be used. Changes to surface conditions and/or geometry resulting from the fire should be recognized in the post-fire assessment as these might affect both radiation and convection heat losses. Allowance should be made for continued heat input if package components would continue to burn following the thermal test exposure.

728.32. Consideration should be given to the proper modelling of any thermal shields such as impact limiters that are affected after the mechanical tests stated in para. 727. Some examples are; changes in shape/dimensions, changes in material densities due to compaction, and separation of the thermal shield.

728.33. Calculations that are performed using finite difference or finite element models should have a sufficiently fine mesh or element distribution to properly represent internal conduction and external and internal boundary conditions. External features such as fins should be given special attention as temperature gradients can be severe, perhaps requiring separate detailed calculations in order to determine the heat flux to the main body. Consideration should be given to the choice of one, two or three dimensional models and to the decision whether the whole package or separate parts are to be evaluated.

728.34. External surfaces of low thermal conductivity can lead to oscillations in computed temperatures. Special techniques (e.g., simplified boundary conditions) or assumptions (e.g., that time-averaged temperatures are sufficiently accurate) might be required to deal with this.

728.35. Generally, conduction and radiation can be modelled explicitly and external convection provides few problems for general purpose computer codes but experimental evidence may be required to support modelling assumptions and basic data used to represent internal convection and radiation. Radiation reflection will be important in gas filled packages and insufficient knowledge of thermal emissivities may restrict the final accuracy. A sensitivity study with different emissivities can be used to show that the assumptions are adequate or to provide conservative (i.e., maximum) limits on calculated temperatures.

728.36. Internal convection will be important for a water filled package and might be significant in a gas filled package. This process is difficult to predict unless there is experimental evidence to support modelling assumptions. Where water circulation routes are provided, internal heat dissipation will be rapid compared with other time constants and simplifying assumptions may be made (e.g., water can be modelled by an artificial material with a high conductivity). Care should be taken to consider areas not subject to circulation (stagnant regions) as high temperatures can occur there because of the inherently low thermal conductivity of water.

728.37. Gas gaps and contact resistances can vary with the differential expansion of components and it is not always clear whether an assumption will yield high or low temperatures. For example, a high resistance gas gap will prevent heat flow, minimizing temperatures inside but maximizing other temperatures because of the reduced effective heat capacity. In such cases calculations based on two extreme assumptions might provide evidence that both conditions are acceptable and, by implication, all variations in between are also acceptable. The gaps and contact resistance in the test sample should be representative of future production. Seals are rarely represented explicitly, but local temperatures could be used as a close approximation to the temperature of the seals.

728.38. The calculation of a thermal test transient should include representation of the initial conditions, 30 minutes with external conditions representing the fire and the cool-down period extended until all temperatures are decreasing with time. In addition, further calculation runs, perhaps with a different mesh distribution, should be performed to check the validity of the model and to assess the uncertainties associated with the modelling assumptions.

728.39. The results of the analysis will be used to confirm that the package has adequate strength and that leakage rates will be acceptable. The determination of pressures from calculated temperatures is thus an important step, particularly where the package contains a volatile material such as water or UF_6. Items such as lead shields often may not be allowed to melt as the resulting condition cannot be accurately defined and thus shielding assessments may not be possible. Component temperatures, if necessary in connection with local hot spots, should be examined to ensure that melting or other modes of failure will not occur. The whole procedure should recognize the uncertainties in the model, the data (e.g., manufacturing tolerances) and the limitations of the computer codes, and allowances made for these uncertainties.

728.40. The post-exposure equilibrium temperatures and pressure might be affected by irreversible changes in the thermal test (perhaps due to protective measures such as the use of expanding coatings or the melting and subsequent relocation of lead within the package). These effects should be assessed.

729.1. As a result of transport accidents near or on a river, lake or sea, a package could be subjected to an external pressure from submersion under water. To simulate the equivalent damage from this low probability event, the Regulations require that a packaging be able to withstand external pressures resulting from submersion at reasonable depths. Engineering estimates indicated that water depths near most bridges, roadways or harbours would be less than 15 m. Consequently, 15 m was selected as the immersion depth for packages (it should be noted that packages containing large quantities of irradiated nuclear fuel should be able to withstand a greater depth (see para. 730)). While immersion at depths greater than 15 m is possible, this value was selected to envelop the equivalent damage from most transport accidents. In addition, the potential consequences of a significant release would be greatest near a coast or in a shallow body of water. The eight hour time period is sufficiently long to allow the package to come to a steady state from rate dependent effects of immersion (e.g., flooding of exterior compartments).

729.2. The water immersion test may be satisfied by immersion of the package, a pressure test of at least 150 kPa, a pressure test on critical components combined with calculations, or by calculations for the whole package. The entire package may not have to be subjected to a pressure test. Justification of model assumptions about the response of critical components should be included in the evaluation.

Enhanced water immersion test for Type B(U) and Type B(M) packages containing more than 10^5 A2 and Type C packages

730.1. See paras 657.1 to 657.8 and 729.1 and 729.2.

730.2. The water immersion test may be satisfied by the immersion of the package, a pressure test of at least 2 MPa, a pressure test on critical components combined with calculations, or by calculations for the whole package.

730.3. If calculational techniques are adopted it should be noted that established methods are usually intended to define material, properties and geometries which will result in a design capable of withstanding the required pressure loading without any impairment. In the case of the 200 m water immersion test requirement for a period of not less than one hour, some degree of buckling or deformation is acceptable provided the final condition conforms with para. 657.

730.4. The entire package does not have to be subjected to a pressure test. Critical components such as the lid area may be subjected to an external gauge pressure of at least 2 MPa and the balance of the structure may be evaluated by calculation.

Water leakage test for packages containing fissile material

732.1. This test is required because water in-leakage may have a large effect on the allowable fissile material content of a package. The sequence of tests is selected to provide conditions which will allow the free ingress of water into the package, together with damage which could rearrange the fissile contents.

733.1. The submersion test is intended to ensure that the criticality assessment is conservative. The sequence of tests prior to the submersion simulate accident conditions that a package could encounter during a severe accident near or on water in transport. The specimen is immersed in at least 0.9 m of water for a period of not less than eight hours.

Tests for Type C packages

734.1. The Regulations do not require the same specimen to be subjected to all the prescribed tests because no real accident sequence combines all the tests at their maximum severity. Instead, the Regulations require the tests to be performed in sequences that concentrate damage in a logical sequence typical of severe accidents, see IAEA TECDOC-702 [130].

734.2. Different specimens may be subjected to the sequences of tests. Also the evaluation criterion for the water immersion test prescribed in para. 730 is different from the criterion for the other tests. The evaluation of the package with regard to shielding and containment integrity must be performed after completing each test sequence.

735.1. The possible occurrence of puncture and tearing is significant. However, the environment is qualitatively and quantitatively difficult to describe [131],[132]. Puncture damage could be caused by parts of the airframe and the cargo. Puncture on the ground is possible but considered to be of less importance.

735.2. A consequence of puncture can be a release from the package containment system, but this would have a very low probability of occurrence. A stronger concern is that of damage to the thermal insulation capability of a package, which would result in unsatisfactory behaviour should a fire follow impact.

735.3. The design of the test requires the definition of a probe with length, diameter, and mass; an unyielding target; and an impact speed. One possibility for specifying the probe was to refer to components of the aircraft. An I-beam has been incorporated in some tests or test proposals, but it was preferred to adopt a more conventional geometric object, namely, a right circular cone. This shape is considered to be one that could cause considerable damage. The height of fall or traveling distance of a probing structure in the range of a few metres is representative of the collapse of structures or bouncing within the aircraft.

735.4. Failure in engines can generate unconfined engine fragments at a rate that deserves consideration. Loss of the aircraft is only one among many possible consequences of the emission of missiles, which can be quite energetic (up to 10^5 J). However, the probability of a fragment hitting a package has been found to be very low in specific studies [130],[133],[134] and penetration probability, although not estimated, would be lower. Thus, on a probability basis, it was considered unnecessary to define a test to cover engine fragment damage.

735.5. For 735(a), the total length of the penetrator probe and details of its construction beyond the frustum are left unspecified but should be adjusted to assure that the mass requirement is attained. For 735(b) the penetrating object should be of sufficient length and mass to extend through the energy-absorbing and thermal-insulating materials surrounding the inner containment vessel; and should be of sufficient rigidity to provide a penetrating force without itself being crushed or collapsed. In both cases, the probe and packaging centers of gravity should be aligned to preclude non-penetrating deflection [135].

735.6. See also para. 727 for additional information.

736.1. The duration of the fire test for air accident qualification was set at 60 minutes. Statistical data on fires resulting from air accidents support the conclusion that the 60-minute thermal test exceeds most severe fire environments that a package would be likely to encounter in an aircraft accident. Fire duration statistics are frequently biased by duration of burning of ground structures and other features not related to the aircraft wreckage, as well as by the location of consignments involved in the accident. To account for this effect, fire-duration information was evaluated carefully to avoid bias by accounts of fires that did not involve the aircraft. The fire test has the same characteristics as those specified in para. 728.

736.2. The importance of fireballs as a severe air accident environment was evaluated in setting the requirements of the fire test. Surveys have shown that “fireballs” of short duration and high temperature occur commonly in the early stages of aircraft fires and are generally followed by a ground fire [136],[137]. The heat input to the package arising from fireballs is not significant compared with the heat input from the extended fire test. Consequently, no tests are required to evaluate a fireball’s impact on package survival.

736.3. The presence of certain materials in an aircraft, for example, magnesium, could result in an intense fire. However, this is not considered to be a serious threat to the package because of the small quantities of such material that are likely to be present and the localized nature of such fires. Similarly, aluminum in large quantities is present in the form of fuselage panels. These panels will have melted away within a few minutes. It was not considered credible that aluminum would burn and increase package heat load greatly.

736.4. This test is not sequential to the 90 m/s impact speed test that is described in para. 737. In severe accidents, high-speed impact and long-duration fires are not expected to be encountered simultaneously because high velocity accidents disperse fuel and lead to non-engulfing, wider area fires of lower consequence. The Type C package must be subjected to an extended test sequence consisting of: the Type B(U)/Type B(M) impact and crush tests (paras. 727(a) and 727(c)), followed by the puncture/tearing test (para. 735) and completed by the enhanced thermal test (para. 736). It is considered that the additive combination of these tests provides protection against severe air accidents that could involve both impact and fire.

736.5. Account should be taken of melting, burning, or other loss of the thermal insulant or structural material upon which the insulant depends for its effectiveness in the longer duration of this fire compared with that for Type B(U) and Type B(M) packages.

736.6. For further material see also para. 728.

737.1. In determining the conditions for the test, the goal was to define the combination of specified velocities normal to an unyielding target that will produce damage conditions to the specimen equivalent to those that might be expected from aircraft impacts at actual speeds onto real surfaces and at randomly occurring angles. Probabilistic distributions of the variable in accidents were considered as well as the package orientation that is most vulnerable to damage.

737.2. Data on which to base accident analyses have been obtained from reports on the particulars of accidents that are filed by officials on the scene and those involved in subsequent investigations. Some of the data are based on actual measurements. Other data are derived by analysis of data and inferences based on a notion of how the accident probably progressed. Each accident report must be evaluated and converted to some basic characteristics, such as impact speed, character of the impacted mass, impact angle, nature of the impact surface, and the like. It is frequently necessary to obtain other accounts of an accident to cross-check information.

737.3. Basic data that might come from an accident report are useful, but do not include the effects of the character of the accident or the environment likely to have been experienced by the cargo involved. For instance, the damage to conveyance and the cargo could be very different if the conveyance impacted a small car, a soft bank, or a bridge abutment. To account for this effect, an analysis is performed to translate the actual impact velocity into an effective head-on impact velocity onto a surface that itself absorbs none of the energy of the impact. Such a surface is called an unyielding surface. Thus, all of the available energy ends up in deformation of the conveyance and the cargo of radioactive material packages. Since the analyst is interested in the cargo, it is normal to assume that the conveyance absorbs no energy; this assumption leads to conservative analysis.

737.4. With the assumption that the cargo impacts at the speed of the conveyance, an analytic translation to effective impact speed onto an unyielding surface will result in an effective impact speed that is lower and depends on the relative strength of the cargo compared to that of the actual impacting surface. For a “hard” package and “soft” target (for example, a spent fuel flask on water) the ratio of actual to effective velocity might range from 7 to 9. For similar hardness in package and surface, the ratio might be 2 or more. For concrete roadways and runways, the velocity ratio could range from 1.1. to 1.4. There are very few surfaces for which the ratio would be 1 [130].

737.5. Conversion of the basic accident report data to effective impact velocity is performed to normalize the accident environment for impact in a standard format that removes much of the variability of the accident scenarios but, at the same time, preserves the stress on the cargo. Repeating this process for all relevant aircraft accidents produces a statistical basis for choosing an effective impact speed onto a rigid target [135],[136],[137].

737.6. Package designs that release no more than an A_2 quantity of radioactive material in a week when subjected to performance testing might be assumed to release their total contents at just slightly more severe conditions. However, such eventualities are not expected. Rather it is expected that a package designed to meet the Regulations will limit releases to accepted levels until the accident environments are well beyond those provided in the performance standards and then will only gradually allow increased release as accident environments greatly exceed the performance test levels; that is, packages should "fail gracefully." This behaviour results from:

(1) The factors of safety incorporated into package designs;
(2) The capability of materials used in the package for a specific purpose, such as shielding, to mitigate loads when that capability is not explicitly considered in the design analysis;
(3) Material capability to resist loads well beyond the elastic limit; and
(4) Reluctance of designers to use and/or competent authorities to approve materials that have abrupt failure thresholds as a result of melting or fracturing in environments likely to occur in transport.

737.7. While all of these features of good package design are expected to provide the desired property of graceful failure, it is also true that there are only very limited data available on packages tested to failure to see how release increases with severity of the accident environment. Limited test data and analyses that have been performed support the concept of graceful failure [137],[138],[139].

737.8. The impact velocity for the test was derived from frequency-distribution cumulative probability studies [130],[140],[141],[142]. Most accident environment analyses reveal that, as the severity of the impact environment increases, the number of events with that severity increases rapidly to a peak and then falls to zero as the severity approaches a physical limit, such as the top speed limitations of the conveyance. Plotting these data as a cumulative curve, that is, percent of events with severity less than a given value, gives a curve that rises quickly at first and then rises very slowly after the "knee" of the curve is reached. When the data are plotted in a format that shows the probability of exceeding a given impact velocity, the scarcity of severe accidents manifests as a distinct bend or "knee" in the curve. This area of the curve is of interest because it indicates where increased levels of protection built into a package begin to have less effect on the probability of failure. Furthermore, the area to the left of the "knee" covers approximately 95% of all accidents. The knee of the curve occurs at about 90 m/s. This value was chosen for the normal component for the impact test .

737.9. Requiring a package design to protect against a normal velocity much higher than the knee generally means a more massive, more complicated, and more expensive package design that achieves little increase in the protection afforded the public. In addition, a design that survives impact at the velocity at the knee will survive many accidents at speeds above the knee because of the conservatism contained in the analysis of accident data and the conversion of that data into effective impact speed onto an unyielding target. Moreover, the fact that a package is designed for graceful failure means that, at these extremes of the curves, the total consequence of packaging failure will be much less than if there were immediate and complete catastrophic failure of containment .

737.10. The need for a package terminal velocity test was discussed in context of the impact test, but it is expected that the impact of a package at terminal velocity is taken into account by the 90 m/s impact test. The purpose of a terminal velocity condition would be to demonstrate that the package design would provide protection in the event that the package is ejected overboard from the aircraft. This situation could arise as a result of mid-air collision or in-flight airframe failure. Nevertheless, it is noted that Type C package requirements already include an impact test on an unyielding surface at a velocity of 90 m/s. This test provides a rigorous demonstration of package integrity for cargo overboard scenarios.

737.11. While the free-fall package velocity may exceed 90 m/s, it is unlikely that the impact surface would be as hard as the unyielding surface specified in the impact test. It is also noted that the probability of aircraft accidents of any type is low and that the percentage of such accidents that involve mid-air collisions or in-flight airframe failures is very low. If such an accident were to occur to an aircraft carrying a Type C package, damage to the package could be mitigated if the package remained attached to airframe wreckage during descent, which would tend to reduce the package impact velocity .

737.12. Subjecting a package to an impact on an unyielding surface with an impact speed of 90 m/s is a difficult test to perform well. This impact speed corresponds to a free drop through a height of about 420 m, without taking into consideration air resistance. This means that guide wires will generally be needed to assure that the package impacts in the desired spot and with the correct orientation. Guided free fall will mean that friction must be accounted for in an even greater release height to assure the speed at impact is correct. Techniques that utilize additional sources of energy to achieve speed and orientation reliability may also be used. These techniques include rocket sleds and cable pulldown facilities.

737.13. Additionally, useful information is provided in paras. 701.1 to 701.24 and in paras 727.6 to 727.17.

737.14. For a package containing fissile material in quantities not excepted by para. 672, the term "maximum damage" should be taken as the damaged condition that will result in the maximum neutron multiplication factor.

SECTION VIII

APPROVAL AND ADMINISTRATIVE REQUIREMENTS

GENERAL

801.1. The Regulations distinguish between cases where the transport can be made without competent authority package design approval and cases where some kind of approval is required. In both cases the Regulations place primary responsibility for compliance on the consignor and the carrier. The consignor should be able to provide documentation in order to demonstrate to the competent authority e.g., by calculations or by test reports, that the package design fulfills the requirements of the Regulations.

801.2. The relevant competent authority may also include competent authorities of countries en route.

802.1. See paras 204.1 to 204.4 and 205.1.

802.2. In the case where competent authority approval is required, an independent assessment by the competent authority should be undertaken, as appropriate, in respect of special form or low dispersible radioactive material; packages containing 0.1 kg or more of UF_6; packages containing fissile materials; Type B(U) and Type B(M) packages; Type C packages; special arrangements; certain shipments; radiation protection programmes for special use vessels; and the calculation of unlisted A_1 and A_2 values, unlisted activity concentrations for exempt material, and unlisted activity limits for exempt consignments.

802.3. Regarding the requirement for competent authority approval for packages designed to contain fissile material, it is noted that para. 672 excludes certain packages from those requirements that apply specifically to fissile material. However, all relevant requirements that apply to the radioactive, non-fissile, properties of the package contents still apply.

802.4. The relationship between the competent authority and the applicant has to be clearly understood. It is the applicant's responsibility to `make the case' to demonstrate compliance with the applicable requirements. The competent authority's responsibility is to judge whether or not the information submitted adequately demonstrates such compliance. It should be free to check statements, calculations and assessments made by the applicant, even, if necessary, by performance of independent calculations or tests. However, the competent authority should not `make the case' for the applicant, because this would put the competent authority in the difficult position of being both `advocate' and `judge'. Nevertheless, this does not prohibit it from providing informal advice to the applicant, without commitment, as to what is likely to be an acceptable way of demonstrating compliance.

802.5. Further details of the role of the competent authority can be found in regulations issued nationally or by the international transport organizations.

802.6. The applicant should contact the competent authority during the preliminary design stage to discuss the implementation of the relevant design principles and to establish both the approval procedure and the actions which should be carried out.

802.7. Experience has shown that many applicants make their first submission in terms of a specific and immediate need which is rather narrow in scope, and then later make several requests for amendments to the approval certificate as they attempt to expand its scope to use the packaging for other types of material and/or shipment. Whenever possible, applicants should be encouraged to make their first submission a general case, which will anticipate and cover their future needs. This will make the `application-approval' system operate more efficiently. Additionally, in some cases, it is mutually advantageous for the prospective applicant and the competent authority to discuss a proposed application in outline before it is formally submitted in detail.

802.8. Further guidance is given in Annex II of IAEA Safety Series No. 112 [12].

APPROVAL OF SPECIAL FORM RADIOACTIVE MATERIAL AND LOW DISPERSIBLE RADIOACTIVE MATERIAL

803.1. The design of special form radioactive material is required to receive unilateral competent authority approval prior to transport, while the design for low dispersible material requires multilateral approval. Para. 803 specifies the minimum information to be included in an application for approval.

804.1. Detailed advice on identification marks is given in para. 828.1 to 828.3.

APPROVAL OF PACKAGE DESIGNS

Approval of package designs to contain uranium hexafluoride

805.1. The approval of packages designed to carry non-fissile or fissile excepted uranium hexafluoride in quantities greater than 0.1 kg is a new requirement, introduced in the 1996 edition of the Regulations. Because this edition of the Regulations introduced specific design and testing requirements, it became necessary to require certification. Thus, a new category of package identification was introduced (see para. 828), and certification of package designs requiring multilateral approval will be required three years earlier than will certification of unilaterally approved package designs. This step was taken to ensure that those designs which do not satisfy all of the new requirements are addressed early in the certification process.

Approval of Type B(M) package designs

810.1. Information given by the applicant with regard to subparas 810(a) and 810(b) will enable the competent authority to assess the implications of the lack of conformance of the Type B(M) design with Type B(U) requirements as well as to determine whether the proposed supplementary controls are sufficient to provide a comparable level of safety. The purpose of supplementary controls is to compensate for the safety measures that could not be incorporated into the design. Through the mechanism of multilateral approval the design of a Type B(M) package is independently assessed by competent authorities in all countries through or into which such packages are transported.

810.2. Special attention should be given to stating which of the Type B(U) requirements of paras 637, 653, 654 and 657-664 are not met by the package design. Proposed supplementary operational controls or restrictions (i.e., other than those already required by the Regulations) which are to be applied to compensate for failure to meet the above mentioned requirements should be fully identified, described, and justified. The maximum and minimum ambient conditions of temperature and solar insolation which are expected during transport should be identified and justified with reference to the regions or countries of use and appropriate meteorological data. See also 665.1 and 665.2.

810.3. Where intermittent venting of Type B(M) packages is required a complete description of the procedures and controls should be submitted to the competent authority for approval. Further advice may be found in paras 666.1 to 666.6.

Approval of package designs to contain fissile material

812.1. Multilateral approval is required for all package designs for fissile material (IF, AF, B(U)F, B(M)F and CF) primarily because of the nature of the criticality hazard and the importance of maintaining subcriticality at all times in transport. Moreover, the regulatory provisions for package design for fissile materials allow complete freedom as to the methods, usually computational, by which compliance is demonstrated. It is therefore necessary that competent authorities independently assess and approve all package designs for fissile materials.

812.2. A package design for fissile material is required to meet the requirements regarding both the radioactive and fissile properties of the package contents. Regarding the radioactive properties, a package is classified in accordance with the definition of package in para. 230. As applicable, a package design approval based on the radioactive, non-fissile properties of the package contents is required. In addition to such approval, a design approval is required relating to the fissile properties of the package contents. See para. 672 for exceptions regarding requirements on package design approval for fissile material.

813.1. The information provided to the competent authority with the application for approval is required to detail the demonstration of compliance with each requirement of paras 671, and 673-682. In particular, the information should include the items specifically quoted in the competent authority approval certificate as detailed in para. 833(m). The inclusion of appropriate information on any experiments, calculations or reasoned arguments used to demonstrate the subcriticality of the individual package or of arrays of packages is acceptable. Sufficient information should be submitted to permit the competent authority to verify compliance of the package with these regulations.

TRANSITIONAL ARRANGEMENTS

Packages not requiring competent authority approval of design under the 1985 and 1985 (As Amended 1990) Editions of these Regulations

815.1. Following from the adoption of the 1985 Editions of the Regulations, packages not requiring approval of design by competent authority based on the 1973 Edition of the Regulations and the 1973 (As Amended) Edition of the Regulations could no longer be used. Continued operational use of such packages required either that the design be reviewed according to the requirements of the 1985 Editions of the Regulations, or that shipments be reviewed and approved by the competent authority as special arrangements, although this was not explicitly stated in the Regulations.

815.2. Paragraph 815 was introduced into the 1996 Edition of the Regulations to allow such existing packagings to continue in use for a limited and defined period of time, following publication, during which the designs might be reviewed, and if necessary modified, to ensure they meet the requirements of the 1996 Edition of the Regulations in full. Where such review and/or modification proves impractical, the transition period allows time for packagings to be phased out and new designs meeting the requirements of the 1996 Edition of the Regulations to be phased in. Packages prepared in accordance with the 1985 or 1985 (As Amended 1990) Edition of the Regulations are sometimes stored for many years prior to further shipment. This may be particularly applicable in the case of Industrial or Type A packages containing radioactive waste and awaiting shipment to intermediate or final storage repositories. Paragraph 815 allows such packages, prepared during a defined period of time and when properly maintained, to be transported in the future on the basis of compliance with the 1985 Editions of the Regulations.

815.3. Paragraph 815 emphasizes the requirement to apply quality assurance measures, according to the 1996 Edition of the Regulations, to ensure that only such packages remain in use, where they continue to meet the original design intent or regulatory requirements. This can best be achieved by ensuring that the latest quality assurance measures are applied to post-manufacturing activities such as servicing, maintenance, modification and use of such packages.

815.4. The reference to Section IV of the 1996 Regulations is included to ensure that only the most recent radiological data (as reflected in A_1 and A_2 values) are used to determine package content and other related limits. It should be noted that the scope of the transitional arrangements of the regulations only extends to the requirements for certain packagings and packages. In all other aspects e.g., concerning general provisions; the requirements and controls for transport including consignment and conveyance limits; and approval and administrative requirements, the provisions of the 1996 Edition of the Regulations apply.

815.5. Any revision to the original package design, or increase in contained activity, or addition of other types of radioactive materials, which would significantly and detrimentally affect safety, as determined by the package owner in consultation with the package designer, will require the design to be reassessed according to the 1996 Edition of the Regulations. This could include such things as an increase in the mass of the contents, changes to the closure, changes to any impact limiters, changes to the thermal protection and shielding and changes in the form of the contents.

Packages approved under the 1973, 1973 (As Amended), 1985 and 1985 (As Amended 1990) Editions of these Regulations

816.1. Following from the adoption of the 1985 Editions of the Regulations, packages requiring approval of design by competent authority (Type B, Type B(U), Type B(M) packages and package designs for fissile material) based on the 1967 Edition, the 1973 Edition and the 1973 (As Amended) Edition of the Regulations were permitted to continue in use, subject to certain limitations on new manufacture, additional requirements to mark such packages with serial numbers and multilateral approval of all such designs. This provision, known colloquially as "grandfathering", was newly introduced into the 1985 Editions of the Regulations to ease the transition to those Regulations. This allowed packages, provided they were properly maintained and continued to meet their original design intent, to continue in use to the end of their useful design lives. It also provided for a period of time, following publication, during which the designs could be reviewed, and if necessary modified, to ensure packages met the requirements of the 1985 Edition of the Regulations in full. Where such review and/or modification proved impractical, the transition period allowed time for packages to be phased out and new designs meeting the requirements of the 1985 Edition of the Regulations to be phased in.

816.2. The reference to Section IV and para. 680 of the 1996 Regulations are included to ensure that only the most recent radiological data (as reflected in the A_1 and A_2 values), and requirements for fissile material by air, may be used to determine package content and other related limits. It should be noted that the scope of the transitional arrangements of the regulations only extends to the requirements for certain packagings and packages. In all other aspects e.g., concerning general provisions; the requirements and controls for transport including consignment and conveyance limits; and approval and administrative requirements, the provisions of the 1996 Edition of the Regulations apply.

816.3. In the process of developing the 1996 Edition of the Regulations it was determined that there was no need for an immediate change of the Regulations following their adoption, but that changes aiming at a long term improvement of safety in transport were justified. Therefore it was also decided to accept continued operational use of certain packages designed and approved under the 1973 Editions of the Regulations. The continued use of existing packagings with a 1967 Edition based package design approval was considered to be no longer necessary or justified.

816.4. The continued use of approved packages meeting the 1973 or 1973 (As Amended) Editions of the Regulations is subject to multilateral approval from the date the 1996 Edition of the Regulations enters into force, in order to permit the competent authorities to establish a framework within which continued use may be approved. Additionally, no new manufacture of packagings to such designs is permitted to commence. This transition period has been determined on the basis of an assessment of the time needed to incorporate the 1996 Edition of the Regulations into national and international regulations.

816.5. See para. 538.2.

816.6. Any revision to the original package design, or increase in contained activity, or addition of other types of radioactive material, which would significantly and detrimentally affect safety, as determined by the competent authority, will require the design to be reassessed and approved according to the 1996 Edition of the Regulations. Such factors could include an increase in the mass of the contents, changes to the closure, changes to any impact limiters, changes to the thermal protection or shielding and changes in the form of the contents.

816.7. When applying para. 816, the original competent authority identification mark and design type codes, assigned by the original competent authority of design, should be retained both on the packages and on the competent authority certificates of design approval, notwithstanding that these packages become subject to multilateral approval of design. This means that packages originally designated Type B(U) or Type B(U)F under the 1973 Editions of the Regulations should not be redesignated Type B(M) or Type B(M)F, nor should they be redesignated Type B(M)-96 or Type B(M)F-96, when used under the provisions of para. 816. This is to ensure that such packages can be clearly identified as packages "grandfathered" under the provisions of paragraph 816, having been originally approved under the 1973 Editions of the Regulations.

817.1. See paras 816.1 and 816.2.

817.2. In the process of developing the 1996 Edition of the Regulations it was determined that there was no need for an immediate change of the Regulations following their adoption, but that changes aiming at long term improvement of safety in transport were justified. Therefore it was also decided to accept continued operational use of certain packages designed and approved under the 1985 Editions of the Regulations.

817.3. The continued use of approved packages meeting the 1985 or 1985 (As Amended 1990) Editions of the Regulations is subject to multilateral approval after 31 December 2003, in order to permit the competent authorities to establish a framework within which continued use may be approved. Additionally no new manufacture of such packagings is permitted to commence beyond 31 December 2006. These transition periods have been determined on the basis of an assessment of the time needed to incorporate the 1996 Edition of the Regulations into national and international regulations.

817.4. When applying para. 817, the original competent authority identification mark and design type codes, assigned by the original competent authority of design, should be retained both on the packages and on the competent authority certificates of design approval, notwithstanding that these packages become subject to multilateral approval of design beyond 31 December 2003. This means that packages originally designated Type B(U)-85 or Type B(U)F-85 under the 1985 Editions of the Regulations should not be redesignated Type B(M)-85 or Type B(M)F-85, nor should they be redesignated Type B(M)-96 or Type B(M)F-96, when used under the provisions of para. 817. This is to ensure that such packages can be clearly identified as packages "grandfathered" under the provisions of paragraph 817, having been originally approved under the 1985 Editions of the Regulations.

Special form radioactive material approved under the 1973, 1973 (As Amended), 1985 and 1985 (As Amended 1990) Editions of these Regulations

818.1. Paragraph 818 introduces transitional arrangements for special form radioactive material, the design of which is also subject to competent authority approval. It emphasizes the need to apply quality assurance measures according to the 1996 Edition of the Regulations to ensure that such special form radioactive material remains in use, only where it continues to meet the original design intent or regulatory requirements. This can best be achieved by ensuring that the latest quality assurance measures are applied to post-manufacturing activities such as servicing, maintenance, modification and use of such special form material. It should be noted that the scope of the transitional arrangements of the regulations only extends to the requirements for certain special form radioactive materials. In all other aspects e.g., concerning general provisions; the requirements and controls for transport including consignment and conveyance limits; and approval and administrative requirements, the provisions of the 1996 Edition of the Regulations apply.

818.2. In the process of developing the 1996 Edition of the Regulations it was determined that there was no need for an immediate change of the Regulations following their adoption, but that changes aiming at a long term improvement of safety in transport were justified. Therefore it was also decided to accept continued operational use of special form radioactive material designed and approved under the 1973 or 1985 Editions of the Regulations. However, no new manufacture of such special form radioactive material is permitted to commence beyond 31 December 2003. The continued use of existing special form radioactive material with a 1967 Edition based design approval was considered to be no longer necessary or justified.

NOTIFICATION AND REGISTRATION OF SERIAL NUMBERS

819.1. The competent authority should monitor specific facets associated with the design, manufacture and use of packagings within its compliance assurance programme (see para. 311). In order to verify adequate performance, the serial number of all packagings manufactured to a design approved by a competent authority is required to be made available to the competent authority. The competent authorities should maintain a register of the serial numbers.

819.2. Packagings manufactured to a package design approved for continued use under the `grandfather' provisions in paras 816 and 817 are also to be assigned a serial number. The serial number, and competent authority knowledge of this serial number, is essential in that the number establishes the means to positively identify which single individual packagings are subject to the respective `grandfather' provision.

819.3. The packaging serial number should uniquely identify each packaging manufactured. The appropriate competent authority is to be informed of the serial number. The term `appropriate' has a broad interpretation and could pertain to any of the following:

- the country where the package design originated;
- the country where the packaging was manufactured; or

- the country or countries where the package is used.

In the case of packagings manufactured to a package design approved for continued use under paras 816 and 817, all competent authorities involved in the multilateral approval process should receive information on packaging serial numbers.

APPROVAL OF SHIPMENTS

820.1. Where shipment approvals are required, such approvals must cover the entire movement of a consignment from origin to destination. If the consignment crosses a national border the shipment approval must be multilateral, i.e., the shipment must be approved by the competent authority of the country in which the shipment originates, and by the competent authorities of all the countries through or into which the consignment is transported. The purpose of the requirement of multilateral approval is to enable the competent authorities concerned to judge the need for any special controls to be applied during transport.

820.2. Each requirement in para. 820 should be applied separately. For example, a consignment of a vented Type B(M) package containing fissile material may need a shipment approval according to both paras 820(a) and 820(c).

820.3. The need to apply para. 820 is governed by the actual content of the package to be transported. For example, when a Type B(M) packaging, for which the package design approval certificate gives the permitted contents as ^{60}Co limited to 1600 TBq, is used for shipment of only 400 TBq ^{60}Co, no shipment approval is required since 400 TBq is less than 1000 TBq.

821.1. According to subparas 802(a)(iv), 802(a)(v), and 802(a)(vi) package design approvals are required for defined package designs. Some of those packages may be transported without additional shipment approval, while for others such approval is required (see para. 820). In some cases, an additional shipment approval is required because operational or other controls may be necessary and those controls may be dependent on the actual package contents. In situations where the need for controls during shipment can be determined at the design review and approval stage the need to review single shipments does not exist. In such cases the package design and shipment approvals may be combined into one approval document.

821.2. The Regulations conceptually differentiate between design approvals and shipment approvals. A shipment approval may be incorporated into the corresponding design approval certificate, and if this is done care should be exercised to clearly define the dual nature of the approval certificate and to apply the proper type codes. For type codes see para. 828.

APPROVAL OF SHIPMENTS UNDER SPECIAL ARRANGEMENT

824.1. Although an approval of a shipment under special arrangement will require consideration of both the shipment procedures and the package design, the approval is conceptually a shipment approval. Further guidance may be found in 312.1.-312.4.

825.1. The level of safety necessary in special arrangement shipments is normally achieved by imposing operational controls to compensate for any non-conformances in the packaging or the shipping procedures. Some of the operational controls which may be effectively employed are as follows:

(a) Exclusive use of vehicle (see para. 221).

(b) Escort of shipment. The escort is normally a radiation protection specialist who is equipped with radiation monitoring instruments and is familiar with emergency procedures enabling him in the event of an accident or other abnormal event to identify quickly any radiation and contamination hazards present and to provide appropriate advice to the civil authorities. For road transport the escort, whenever possible, should travel in a separate vehicle so that he would not be incapacitated by the same accident. The escort should also be equipped with stakes, ropes, and signs to cordon off an accident area and with a fire extinguisher to control minor fires, and a communications system. If considered prudent, the radiation protection specialist could be accompanied by police and fire department escorts.

(c) Routeing of shipment may be controlled in order to select the potentially least hazardous routes and, if possible, to avoid areas of high population density and possible hazards, such as steep gradients and railway level crossings.

(d) Timing of shipment may be controlled to avoid busy periods such as rush hours and weekend traffic peaks.

(e) Shipments should be made directly, i.e., without stopover or transshipment, where possible.

(f) Transport vehicle speeds may be limited, particularly if the impact resistance of the packaging is low and if the slower speed of the transport vehicle would not cause additional hazards (such as collisions involving faster moving vehicles).

(g) Consideration should be given to notifying the emergency services (police and fire departments) in advance.

(h) Emergency procedures (either ad hoc or standing) should exist for contingencies resulting from the shipment being involved in an accident.

(i) Ancillary equipment such as package-to-vehicle tie-down or shock absorber systems and other protective devices or structures should be used, where necessary, as compensatory safety arrangements.

COMPETENT AUTHORITY APPROVAL CERTIFICATES

Competent authority identification marks

828.1. In applying and interpreting the type codes it is necessary to keep in mind that the code is based on the use of several indicators intended to quickly provide information on the type of package or shipment in question. The indicators provide information on package design characteristics (e.g., Type B(U), Type B(M)), or Type C), or on the possible presence of fissile material in the package and on other specific aspects of the approval certificate (e.g., special arrangement, shipment, special form, low dispersible radioactive material or non-fissile or fissile-excepted uranium hexafluoride contents). Specifically, the appearance of, for example, B(U)F in the code does not necessarily imply the presence of fissile material in a particular package, only the possibility that it might be present.

828.2. It is essential that easy means are available for determining under which edition of the Regulations the original package design approval was issued, preferably in the identification mark. This will be achieved by adding the symbol `-96' to the type code.

Example:

Edition of Regulations	Package design identification mark
1967 Edition	A/132/B

1973 Edition	A/132/B(U), or A/132/B(M)
1985 Edition	A/132/B(U)-85, or A/132/B(M)-85
1996 Edition	A/132/B(U)-96, or A/132/B(M)-96

828.3. This technique of adding a symbol may continue to be used provided later editions of the Regulations essentially maintain the present package type codes.

CONTENTS OF APPROVAL CERTIFICATES

Special form radioactive material and low dispersible radioactive material approval certificates

830.1. The purpose of the careful description of approval certificate content is twofold. It aims at providing assistance to competent authorities in designing their certificates, and it facilitates any checking of certificates, because the information they contain is standardized.

830.2. The Regulations prescribe the basic information which must appear on certificates of approval and a competent authority identification mark system. Competent authorities are urged to follow these prescriptions as closely as possible to achieve international uniformity of certification. In addition to the applicable national regulations and the relevant international regulations, each certificate should make reference to the appropriate edition of the Regulations, because this is the internationally recognized and known standard. The international vehicle registration (VRI) code, which is used in competent authority identification marks, is given in Table IV.

TABLE IV. LIST OF VRI CODES BY COUNTRY

COUNTRY	VRI Code	COUNTRY	VRI Code
Afghanistan	AFG	Malaysia	MAL
Albania	AL	Mali	RMM
Algeria	DZ	Mauritius	MU*
Argentina	RA	Mexico	MEX
Australia	AUS	Monaco	MC
Austria	A	Mongolia	MN*
Bangladesh	BD	Morocco	MA*
Belgium	B	Myanmar	BUR
Bolivia	BO*	Namibia	NA*
Brazil	BR	Netherlands	NL
Bulgaria	BG	New Zealand	NZ
Cameroon	CM*	Nicaragua	NI*

Canada	CDN	Niger	RN
Chile	RCH	Nigeria	WAN
China, People's Rep. of	CN*	Norway	N
Colombia	CO	Pakistan	PAK
Costa Rica	CR	Panama	PA*
Cote d'Ivoire	CI*	Paraguay	PY
Cuba	C	Peru	PE
Cyprus	CY	Philippines	RP
Czech Republic	CZ	Poland	PL
Democratic Kampuchea	KH*	Portugal	P
Denmark	DK	Qatar	QA*
Dominican Republic	DO*	Romania	R
Ecuador	EC	Russian Federation	RU
Egypt	ET	Saudi Arabia	SA
El Salvador	ES	Senegal	SN*
Ethiopia	ETH	Sierra Leone	WAL
Finland	FIN	Singapore	SGP
France	F	Slovak Republic	SK
Gabon	GA*	Slovenia	SLO
German Dem. Rep. (former)	DDR	South Africa	ZA
Germany	D	Spain	E
Ghana	GH	Sri Lanka	CL
Greece	GR	Sudan	SUD
Guatemala	GCA	Sweden	S
Haiti	HI*	Switzerland	CH
Holy See	VA*	Syrian Arab Republic	SYR
Hungary	H	Thailand	T
Iceland	IS	Tunisia	TN
India	IND	Turkey	TR
Indonesia	RI	Uganda	UG*

Iran, Islamic Republic of	IR	Ukraine	UA
Iraq	IRQ	United Arab Emirates	AE*
Ireland	IRL	United Kingdom	GB
Israel	IL	United Republic of Tanzania	TZ*
Italy	I	United States of America	USA
Jamaica	JA	Uruguay	U
Japan	J	USSR (former Soviet Union)	SU
Jordan	HKJ	Venezuela	YV
Kenya	EAK	Vietnam	VN
Korea, Dem. P.R.	KP	Yugoslavia	YU
Korea, Republic of	ROK	Zaire	ZRE
Kuwait	KW*	Zambia	Z
Lebanon	RL	Zimbabwe	ZB*
Liberia	LR*		
Libyan Arab Jamahiriya	LY*		
Liechtenstein	LI*		
Luxembourg	L		
Madagascar	RM		

* ISO Code where no VRI Code is available

Special arrangement approval certificates

831.1. As discussed in para. 418.1, during preparation of the certificate, care should be taken relative to the authorized quantity, type, and form of the contents of each package because of the potential impact on criticality safety. Any special inspections or tests of the contents to confirm the characteristics of the contents prior to shipment should be specified in the certificate. This is of particular importance for any removable neutron poison or other criticality control feature that will be loaded in the package prior to shipment (see paras 502.4 and 502.5). Where appropriate the criteria which the measurement must satisfy should be specified or referenced in the approval certificate.

831.2. Any special loading arrangement of the packages that should be adhered to or avoided should be noted in the special arrangement certificate.

Shipment approval certificates

832.1. See para. 831.1.

832.2. With this Edition of the Regulations, packages that contain fissile material are excepted from the requirements of paras 673-682 if certain package and consignment requirements are met (see para. 672a). If the packages in the consignment contain fissile material that is excepted based on the package limits, care should be taken to assure that the consignment limit is not exceeded. This will mean that the consignor should be knowledgable relative to the upper limit of the fissile material quantity in each package or assume that the upper limit (see para. 672a) is contained in each package.

Package design approval certificates

833.1. As discussed in para. 418.1, care should be taken relative to the authorized quantity, type, and form of the contents of each package because of the potential impact on criticality safety. Any inspections or tests of the contents that may be needed to confirm the characteristics of contents prior to shipment should be specified in the certificate. Measurements that satisfy the requirements of subpara. 674(b) may need to be performed prior to loading and/or shipment if the package contains irradiated nuclear fuel. The criteria that the measurement must satisfy should be specified or referenced in the certificate for the package. See related advisory material of para. 502.8. Similarly, if special features are allowed to exclude water inleakage, specific inspections and/or test procedures to assure compliance should be stated (or referenced) in the certificate.

VALIDATION OF CERTIFICATES

834.1. The approval certificate of the competent authority of the country of origin is usually the first to be issued in the series of multilateral approval certificates. Competent authorities, other than that of the country of origin, have the option of either performing a separate safety assessment and evaluation or making use of the assessment already done by the original competent authority, thus limiting the scope and extent of their own assessment.

834.2. Subsequent approval certificates may take either of two forms. First, a competent authority in a subsequent country may endorse the original certificate, i.e., agree with and endorse the original certificate including any definition of controls incorporated in it. This is multilateral approval by validation of the original certificate. An approval by validation will not require any additional competent authority's identification mark, either in terms of certificate identification or marking on packages. Second, a competent authority may issue an approval certificate which is associated with, but separate from, the original certificate in that this subsequent certificate would bear an identification mark other than that of the original identification mark. Furthermore, in this case packagings in use under such a multilateral approval have to be marked with the identification marks of both the original and the subsequent approval certificates, (see subpara. 829(b)).

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERTRAN: A System for Assessing the Impact from Transporting Radioactive Material, IAEA-TECDOC-287, IAEA, Vienna (1983).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. IAEA Safety Series No. 115. (Vienna 1996).

[3] AMERICAN NATIONAL STANDARD for Nuclear Criticality Control of Special Actinide Elements, ANSI/ANS-8.15-1981 (Reaffirmed 1987), American Nuclear Society (1981).

[4] INTERNATIONAL ORGANIZATION FOR STANDARDIZATION, Series 1 Freight Containers - Specifications and Testing - Part 1: General Cargo Containers, (ISO 1496:1-1990)(E), ISO, Geneva (1990).

[5] INTERNATIONAL MARITIME ORGANIZATION, International Convention for Safe Containers, IMO, London (1984).

[6] INTERNATIONAL MARITIME ORGANIZATION , International Maritime Dangerous Goods (IMDG) Code.

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[8] INTERNATIONAL ATOMIC ENERGY AGENCY, "Interim guidance for the safe transport of reprocessed uranium", IAEA-TECDOC-750, IAEA, Vienna, June 1994.

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, "Discussion of and Guidance on the Optimization of Radiation Protection in the Transport of Radioactive Material", IAEA TECDOC 374, Vienna

[10] European Agreement Concerning the International Carriage of Dangerous Goods by Road, (ADR) 1997 Edition, marginals 10315, 71315 and appendix B4.

[11] RIDDER, K., "The Training of Dangerous Goods Drivers in Europe" PATRAM 95. (Proc. Sym. Las Vegas 1995).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Compliance Assurance for the Safe Transport of Radioactive Material, Safety Series No.112, IAEA, Vienna (1994).

[13] WILSON C. K., SHAW K. B., GELDER R., "Towards the implementation of ALARA in Transport." Packaging and Transportation of Radioactive Materials. PATRAM '92 (Proc. Sym. Yokohama City 1992).

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